ENCLOSURE 2
SUMMARY EXPLANATION

This submittal includes the transmittal letter and five enclosures. The enclosures include a table of contents (Enclosure 1), this executive summary (Enclosure 2), documentation of state and local government officials agreements (Enclosure 3), detailed justifications for each emergency action level (EAL) (Enclosure 4), and electronic copies of enclosures, references, and supporting information (Enclosure 5).

The Site Emergency Plan for the Palisades Nuclear Plant (PNP) currently uses the NUREG-0654 EAL scheme. Nuclear Management Company, LLC (NMC) requests approval to change the existing scheme for PNP to that described in NEI 99-01, Revision 4, "Methodology for Development of Emergency Action Levels," January 2003, as endorsed by the Nuclear Regulatory Commission (NRC) in Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," Revision 4, July 2003.

In accordance with 10 CFR 50, Appendix E, state and local government officials reviewed the EAL changes, including areas where conversion from NUREG-0654 to NEI 99-01, Revision 4, resulted in a higher or lower classification for a specific event. The following state and local government officials' agreement is contained in Enclosure 3:

Anthony Katarsky, Michigan State Police, Emergency Management Division
Alain Svilpe, Office of Domestic Preparedness, VanBuren County
Michael Bradley, Emergency Preparedness, Berrien County
Scott Corbin, Emergency Preparedness, Allegan County

The detailed supporting information for the proposed EAL changes is contained in Enclosure 4. There are four attachments within Enclosure 4. Attachment 1 contains a red-line, highlighted copy of the Technical Basis Document. This document includes the pertinent information to describe each EAL (category, description, basis, etc). The red-line and highlighted areas indicate changes made to the information contained in NEI 99-01, Revision 4, in order to develop site-specific EALs. Attachment 2 contains a clean copy of the Technical Basis Document. Attachment 3 contains the detailed justification matrix. This matrix provides the cross-reference comparing the current NEI 99-01, Revision 4 EALs, to the proposed EALs, mode applicability, and specific identification and discussion of differences and deviations. Attachment 4 contains the Emergency Plan changes, as depicted on wall charts.

The Technical Basis Document, justification matrix, and supporting reference material are contained on compact disks in Enclosure 5.

<u>Summary of Matrices</u>

The matrices contain justifications for site-specific information, differences and deviations identified in the initiating conditions, EAL thresholds, and basis documents. The matrices are divided into sections following the format of the Technical Basis Document. A section identifying generic differences is at the front of the document.

Overview of Deviations

1) PNP's reactor vessel level monitoring system (RVLMS) design is unable to detect levels that specifically match the NEI levels. Based on the indications available, PNP has applied the NEI methodology to allow indications for each level of classification. However, the levels that can be detected result in conservative classification calls. This affects EALs CA1.1, CA2.1, CS1.1, CS1.2, CS2.1, CS2.2, CG1.2, FPB FC PL#4, FPB CMT PL#3.

2) The NEI guidance for RA1.2 is a release that exceeds 200 times the ODCM. Application of 200 times the ODCM Limit for the Radioactive Gaseous Effluent Monitors (RGEM) listed in RA1.2 (RIA-2326 and RIA-2327) results in a value that exceeds the offsite dose limits specified for RS1.1. Therefore, if the 200 times ODCM limit were used, an Alert would never be declared since the SAE threshold would already be exceeded. To maintain intervals between classifications and declare the Alert at an appropriate level, PNP selected limits for these monitors, which are less than 200 times the ODCM Limits.

3) The basis for SU4 references use of the failed fuel monitor to provide indication of fuel clad integrity. The failed fuel monitor at PNP has been abandoned-in-place, and therefore, an alternate method of detection has been developed for SU4.1.

Overview of Differences

Significant differences identified in the matrices include the following:

1) EALs associated with plant equipment not on site were eliminated. This difference is primarily associated with offsite perimeter monitoring and real time dose assessment. This affects EALs RU1.4, RU1.5, RA1.4, RA1.5, RS1.3, and RG1.3.

2) PNP uses two indications for reactor vessel inventory indication (containment sump OR primary system drain tank). The OR statement was used instead of the NEI wording "and" because there would not be spontaneous indication in both locations. This affects EALs CU2.2, CA1.2, CA2.2, CS1.1, CS1.2, and CG1.1.

3) PNP used bus indication in the classification for loss of offsite power. The PNP wording focuses the classification on the loss of offsite power capability to the essential buses. This affects EALs CU3.1, CA3.1, SU1.1, SS1.1, and SG1.1.

4) PNP is a CE plant and does not use critical safety function status trees (CSFSTs). Therefore, in accordance with the guidance in NEI 99-01, Revision 4, PNP does not have EALs for critical safety function status under the fuel cladding, primary coolant system, and containment barriers in the fission product barrier matrix.

Minor differences identified in the matrices include the following:

Additional minor wording changes have been identified in the generic differences section, as well as in the justification section, for the applicable EAL. These differences do not alter the meaning or intent of the EALs.

In summary, this submittal provides the basis and justification for changing the PNP EAL scheme from the NUREG-0654 requirements to the NEI 99-01 requirements, and demonstrates compliance with 10 CFR 50.54(q).

ENCLOSURE 3

STATE AND LOCAL GOVERNMENT OFFICIAL AGREEMENT DOCUMENTATION

32 Pages Follow

GOVERNMENTAL AGENCY REVIEWS
NEI-99-01 SUBMITAL FOR PALISADES

Palisades Emergency Planning section conducted a review/discussion of the proposed changes to the emergency action levels (EALs) for the Palisades Nuclear Plant on October 6, 2004. The following personnel attended this review/discussion and agree with the proposed change to NEI-99-01 rev.4 based EALs.

State of Michigan, Emergency Management Division

Signed: _____ Date: 10/6/04

VanBuren County Office of Domestic Preparedness

Signed: _____ Date: 10/6/04

Allegan County Emergency Management

Signed: _____ Date: 10-7-04

Berrien County Emergency Management

Signed: _____ Date: 10-6-04

Major Differences in New EALs

BIG PICTURE

- NRC endorsed the new EAL scheme in NEI 99-01, Rev. 4 via Reg. Guide 1.101, Rev. 4, July 2003
- New IC/EAL Scheme added generic:
  - Shutdown and Refueling IC/EALs
  - ISFSI IC/EALs
- New generic approach to Fission Product Barrier IC/EALs
- The PNP EALs are divided into five broad groups:
  - EAL Group **C** – EALs that are related to system or equipment **malfunctions** for cold shutdown.
  - EAL Group **S** – EALs that are related to system or equipment **malfunctions** for all other modes
  - EAL Group **R** – EALs that are **radiological** in nature
  - EAL Group **H** – EALs that are related to external **hazards** such as security events, fires or natural events
  - EAL Group **E** – EALs associated with the Independent Spent Fuel Storage Installation **(ISFSI)**
  - EAL Group **F** – EALs that are related to loss or challenge of one or more **fission product barriers**
- New IC/EAL Scheme is plant operating Mode Specific (Symptom Based)
  - Power Operation -- critical and >5% power.
  - Startup -- critical and ≤5% power.
  - Hot Standby – Reactor Coolant Tavg ≥300°F.
  - Hot Shutdown – Reactor Coolant 300°F > Tavg > 200°F
  - Cold Shutdown -- Reactor Coolant Tavg ≤200°F
  - Refueling -- One or more reactor vessel head closure bolts less than fully tensioned.
- Current EALs are presented as Categories
  - Example: Category – Engineered Safety Features

Major Differences in New EALs

REVIEW OF ICS/EALS

Current EALs Not in New Scheme

- Safety Injection initiated and discharged to vessel. UE
- Failure of a safety or relief valve in a safety related system to close following reduction of applicable pressure. UE
- Main Steam Line/Main Feedwater Line break outside Containment which is isolated by Main Steam isolation signal (manually or auto). UE
- Main Steam Line/Main Feedwater Line break inside or outside Containment which is not isolated. ALERT
- Any challenge to Over-Pressure Protection System (LTOP). UE
- Pressurizer code safety valve operation. UE
- Steam line break with >10gpm primary to secondary leak rate. ALERT (but may be classified under FPB)
- Rapid gross failure of one Steam Generator tubes with loss of offsite power. ALERT

Fission Product Barrier Table

Abnormal Rad/Level Rad Effluent

EAL#	EAL	Category	EAL	Comments
IC RU1	Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Offsite Dose Calculation Manual (ODCM) Specifications for 60 Minutes or Longer	Releases		One to one corresponding category to IC
RU1.1	Valid reading on effluent monitor RIA-1049 "Liquid Radwaste Discharge Monitor" that exceeds two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.	Releases	**UE** Short-term radiological effluent ODCM limits exceeded	Current EAL scheme would classify at lower threshold. New scheme is based on dose criteria.
RU1.2	VALID reading on one or more of the following radiation monitors that exceeds the reading shown for 60 minutes or longer: Stack Normal Range Monitor (RIA-2326) 3.2E+5 cpm S/G Blowdown Monitor (RIA-0707) 2 X High Alarm Service Water Monitor (RIA-0833) 2 X High Alarm Turbine Bldg. Sumps Monitor (RIA-5211) 2 X High Alarm	Releases	**UE** Short-term radiological effluent ODCM limits exceeded.	Current EAL scheme would classify at lower threshold. New scheme is based on dose criteria.
RU1.3	Confirmed sample analysis for gaseous or liquid releases indicates concentrations or release rates, with a release duration of 60 minutes or longer, in excess of two times the Offsite Dose Calculation Manual (ODCM) limit	Releases	**UE** Significant solid or liquid waste spill outside restricted areas with threatened offsite release.	New EAL gives a specific value for releases (2X ODCM).

IC RU2	Unexpected Rise in Plant Radiation	FPB/FD		One to one corresponding category to IC
RU2.1	VALID indication of uncontrolled water level lowering LESS THAN 647 ft el. in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water **AND** UNPLANNED VALID Area Radiation Monitor reading rise on any of the following: Vent Monitor Fuel Handling Area (RIA-5712) Spent Fuel Pool Area Radiation Monitors (RIA-5709 or RIA-2313) Refueling Containment High Radiation (CHR) Monitors (RIA-2316 or RIA-2317)	FPB/FD	**ALERT** Fuel damage/handling accident with release of radioactivity to Containment or Fuel Handling Building.	New EAL provides more specific guidance.
RU2.2	Any UNPLANNED VALID Area Radiation Monitor reading rises by a factor of 1000 over normal* levels. *Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.	Radiation Levels	**ALERT** Radiation levels or airborne contamination indicates a severe degradation in control of radioactive materials.	New EAL provides more specific guidance.
IC RA1	Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Offsite Dose Calculation Manual (ODCM) Specifications for 15 Minutes or Longer	Releases None		"Releases" is the main category for this IC. Current EAL scheme does not directly address release at 200X the release limit.
RA1.1	VALID reading on effluent monitor RIA-1049 "Liquid Radwaste Discharge Monitor" that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer.	Releases	**ALERT** Radiological effluent > 10 times the ODCM instantaneous release rate limit.	Current EAL scheme would classify at lower threshold. New scheme is based on dose criteria.

RA1.2	VALID reading on one or more of the following radiation monitors that exceeds the reading shown for 15 minutes or longer: Stack Normal Range Monitor (RIA-2326) 1.3E+6 cpm Stack High Range Monitor (RIA-2327) 1.5 mRem/hr S/G Blowdown Monitor (RIA-0707) 200 X High Alarm Service Water Monitor (RIA-0833) 200 X High Alarm Turbine Bldg. Sumps Monitor (RIA-5211) 200 X High Alarm	Releases	**ALERT** Radiological effluent > 10 times the ODCM instantaneous release rate limit.	Current EAL scheme would classify at lower threshold. New scheme is based on dose criteria.
RA1.3	Confirmed sample analysis for gaseous or liquid release indicates concentrations or release rates, with a release duration of 15 minutes or longer, in excess of 200 times ODCM limit.	None		No corresponding EAL in existing scheme
IC RA2	Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel	FPB/FD		One to one corresponding category to IC
RA2.1	A VALID alarm or reading on one or more of the following radiation monitors: Vent Monitor Fuel Handling Area RIA-5712 1E+4 cpm Spent Fuel Pool Area Radiation Monitors RIA-5709 or RIA-2313 15 mR/hr Refueling Containment High Radiation (CHR) Monitors RIA-2316 or RIA-2317 80 mR/hr above background	FPB/FD	**ALERT** Fuel damage/handling accident with release of radioactivity to Containment or Fuel Handling Building.	New EAL provides more specific guidance.
RA2.2	Water level less than 614 feet elevation for the reactor refueling cavity, spent fuel pool and fuel transfer canal that will result in fuel uncovering.	FPB/FD	**SAE** Major damage to irradiated fuel in Containment or Fuel Handling Building. (eg, large object damages fuel or water loss below fuel level)	Cold shutdown EALs are also used in classifying this event.

IC RA3	Release of Radioactive Material or Rises in Radiation Revels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown	None		No corresponding EAL in existing scheme
RA3.1	VALID radiation monitor readings GREATER THAN 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions: Control Room (RIA-2310) **OR** Central Alarm Station (RIA-2304)	None		No corresponding EAL in existing scheme

RA3.2	Any VALID radiation monitor reading GREATER THAN 3 R/hr in areas requiring infrequent access to maintain plant safety functions (Table H-1).
	
	• TABLE H-1 SAFE SHUTDOWN BUILDINGS
	  • Containment Structure
	  • Auxiliary Building
	  • Turbine Building
	• Screenhouse
		None		No corresponding EAL in existing scheme

IC RS1	Offsite Dose Resulting from an actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the Actual or Projected Duration of the Release	Releases		One to one corresponding category to IC
RS1.1	NOTE: If dose assessment results are available at the time of declaration, the classification should be based on RS1.2 instead of RS1.1.While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated. VALID reading on one or more of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer: Stack High Range Monitor (RIA-2327) 5.2 mRem/hr Main Steam Line Monitors (RIA-2323/RIA-2324)1.70E+3 cpm Stack High Range Effluent Monitor (RIA-2328)# .64 mRem/hr Atmospheric Dump Valve High Range Effluent Monitors (RIA-2328) # .57 mRem/hr	Releases	**SAE** Effluent monitors detect levels corresponding to > 50 mrem/hr TEDE for ½ hour or > 500 mrem/hr TEDE for 2 minutes (or 5 times these levels to the Adult Thyroid CDE) at the site boundary for adverse meteorological conditions. These levels are projected based on other Plant parameters (eg, radiation level in Containment with leak rate appropriate for existing Containment pressure) or are measured in the environs. EPA Protective Action Guidelines (see Emergency Implementing Procedure EI-6.13, "Protective Action Recommendations for Offsite Populations") are projected to be exceeded outside the site boundary.	Current EAL scheme would classify at lower threshold. New scheme is based on dose criteria.
RS1.2	Dose assessment using actual meteorology indicates doses GREATER THAN 100 mRem TEDE or 500 mRem CDE at or beyond the site boundary.	Releases	See Above	See Above
RS1.3	Field survey results indicate closed window dose rates exceeding 100 mRem/hr expected to continue for more than one hour at or beyond the site boundary. **OR** Analyses of field survey samples indicate thyroid CDE of 500 mRem for one hour of inhalation, at or beyond the site boundary	Releases	See Above	See Above

IC RG1	Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology	Releases		One to one corresponding category to IC
RG1.1	NOTE: If dose assessment results are available at the time of declaration, the classification should be based on RG1.2 instead of RG1.1.While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated. VALID reading on any of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer. Stack High Range Monitor (RIA-2327) 52 mRem/hr Main Steam Line Monitors (RIA-2323/RIA-2324)1.70E+4 cpm Stack High Range Effluent Monitor (RIA-2328)# 6.4 mRem/hr Atmospheric Dump Valve High Range Effluent Monitors (RIA-2328) # 5.7 mRem/hr	Releases	**GE** Effluent monitors detect levels corresponding to 1 rem/hr TEDE or 5 rem/hr Adult Thyroid CDE Rate at the site boundary under actual meteorological conditions. These levels are projected based on other Plant parameters (eg, radiation levels in Containment with leak rate appropriate for existing Containment pressure) or are measured in the environs.	Current EAL scheme would classify at lower threshold. New scheme is based on dose criteria.
RG1.2	Dose assessment using actual meteorology indicates doses GREATER THAN 1000 mRem TEDE or 5000 mRem thyroid CDE at or beyond the site boundary	Releases	See above	See Above
RG1.3	Field survey results indicate closed window dose rates exceeding 1000 mRem/hr expected to continue for more than one hour, at or beyond site boundary. **OR** Analyses of field survey samples indicate thyroid CDE of 5000 mRem for one hour of inhalation, at or beyond site boundary	Releases	See above	See Above

Cold Shutdown/Refueling

EAL#	EAL	Category	Current EAL	Comments
IC CU1	PCS Leakage.	PCS Integrity		One to one corresponding category to IC
CU1.1.	Unidentified or pressure boundary leakage GREATER THAN 10 gpm.	PCS Integrity	**UE** PCS leak rate in excess of Technical Specifications, but < 50 gpm. **Or** Primary to secondary leakage rate exceeds Technical Specifications limits of 432 gallons per day primary to secondary LEAKAGE through any one SG but < 50 gpm.	Current EAL scheme would classify at lower threshold. New EAL is based on severity of event.
CU1.2.	Identified leakage GREATER THAN 25 gpm.	PCS Integrity	See Above	See Above
IC CU2	UNPLANNED Loss of PCS Inventory with Irradiated Fuel in the Reactor Vessel.	FPB/FD		One to one corresponding category to IC
CU2.1.	UNPLANNED PCS level lowering below reactor vessel flange for GREATER THAN OR EQUAL TO 15 minutes.	FPB/FD	**ALERT** Fuel damage/handling accident with release of radioactivity to Containment or Fuel Handling Building.	Current EAL scheme would classify at lower threshold. New EAL is based on severity of event. Was ALERT will be UE.
CU2.2.	Loss of reactor vessel inventory as indicated by unexplained Containment Sump level rise or Primary System Drain Tank level rise. **AND** Reactor Vessel level cannot be monitored	FPB/FD	See Above	See Above

IC CU3	Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes	Plant Power - Electrical		One to one corresponding category to IC
CU3.1.	Loss of all offsite power to Vital 2400 VAC busses 1C and 1D for GREATER THAN 15 minutes. **AND** At least 1 emergency generator is supplying power to each Vital 2400 VAC bus.	Plant Power - Electrical	**UE** Loss of offsite AC power **UE** Loss of emergency onsite AC power	New scheme would not classify loss of emergency onsite power.
IC CU4	UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel	Engineered Safety Features		One to one corresponding category to IC
CU4.1.	An UNPLANNED event results in PCS temperature exceeding the Technical Specification cold shutdown temp limit of 200°F	Engineered Safety Features	**UE** Complete loss of any functions needed for cold shutdown.	New EALs are more descriptive.
CU4.2.	Loss of all PCS temperature and Reactor Vessel level indication for GREATER THAN 15 minutes.	Engineered Safety Features	See above	See above
IC CU5	Fuel Cladding Degradation	FPB/FD		One to one corresponding category to IC
CU5.1.	Coolant sample activity GREATER THAN 40 µCi/gm dose equivalent I-131indicating fuel clad degradation.	FPB/FD	**UE** Primary Coolant Iodine-131 dose concentration equivalent > 1.0μCi/gm for more than 72 hrs **ALERT** Primary Coolant Iodine-131 dose concentration equivalent >32μCi/gm OR > 5% total failed fuel.	Higher limit is based on plant analysis, and 72 hours was removed.

IC CU6	UNPLANNED Loss of All Onsite or Offsite Communications Capabilities	Comm. Loss		One to one corresponding category to IC
CU6.1.	Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations.	Comm. Loss	**UE** Significant loss of offsite communication capability. Loss of the Emergency Notification System (ENS), and all other phones including satellite phones, that could be used to make notifications to Van Buren County, the State of Michigan and the NRC.	Only one EAL in the current scheme. We could classify some events under the new scheme as UEs that would not have been classified at all under the current scheme
CU6.2.	Loss of all Table C-2 offsite communications capability	Comm. Loss	See above	See above
IC CU7	UNPLANNED Loss of Required DC Power for GREATER THAN 15 minutes	Plant Power - Electrical		One to one corresponding category to IC
CU7.1.	UNPLANNED Loss of Vital DC power to required DC busses based on LESS THAN 105 VDC bus voltage indications. **AND** Failure to restore power to at least one required DC bus within 15 minutes from the time of loss	Plant Power - Electrical	**ALERT** Loss of all onsite DC power for less than 15 minutes. **SAE** Loss of all vital onsite DC power for more than 15 minutes, AND Plant not in cold shutdown.	Current scheme would classify at lower threshold without regard to plant status. New scheme takes into account plant at cold shutdown and it takes longer for events to affect the public.
IC CU8	Inadvertent Criticality.	None		No corresponding category in existing scheme
CU8.1.	An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.	None		No corresponding EAL in existing scheme

IC CA1	Loss of PCS Inventory.	FPB/FD None		One to one corresponding category to IC
CA1.1.	Loss of PCS inventory as indicated by Reactor Vessel LESS THAN 617 ft. el.	None		No corresponding EAL in existing scheme
CA1.2.	Loss of PCS inventory as indicated by unexplained Containment Sump or Primary System Drain Tank level rise **AND** Reactor Vessel level cannot be monitored for GREATER THAN 15 minutes	FPB/FD	**SAE** Major damage to irradiated fuel in Containment or Fuel Handling Building. (eg, large object damages fuel or water loss below fuel level)	Current scheme would classify at lower threshold without regard to plant status. New scheme takes into account plant at cold shutdown and it takes longer for events to affect the public.
IC CA2	Loss of Reactor Vessel Inventory with Irradiated Fuel in the Reactor Vessel	FPB/FD		One to one corresponding category to IC
CA2.1.	Loss of Reactor Vessel inventory as indicated by Reactor Vessel LESS THAN 617 ft. el.	None		No corresponding EAL in existing scheme
CA2.2.	Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump or Primary System Drain Tank level rise **AND** Reactor Vessel level cannot be monitored for GREATER THAN 15 minutes	FPB/FD	**SAE** Major damage to irradiated fuel in Containment or Fuel Handling Building. (eg, large object damages fuel or water loss below fuel level)	Current scheme would classify at lower threshold without regard to plant status. New scheme takes into account plant at cold shutdown and it takes longer for events to affect the public.
CA3	Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses.	Plant Power - Electrical		One to one corresponding category to IC
CA3.1.	Loss of all offsite power to both Vital 2400 VAC buses 1C and 1D **AND** Failure of all emergency generators to supply power to Vital 2400 VAC busses. **AND** Failure to restore power to at least one Vital 2400 VAC bus within 15 minutes from the time of loss of both offsite and onsite AC power.	Plant Power - Electrical	**ALERT** Loss of offsite AND onsite AC power for less than 15 minutes. **SAE** Loss of offsite AND onsite AC power for more than 15 minutes.	Current scheme would classify at lower threshold without regard to plant status. New scheme takes into account plant at cold shutdown and it takes longer for events to affect the public.

IC CA4	Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel	FPB/FD		One to one corresponding category to IC
CA4.1.	With CONTAINMENT CLOSURE and PCS integrity not established an UNPLANNED event results in PCS temperature exceeding the Technical Specification cold shutdown temperature limit of 200°F.	FPB/FD	**GE** Loss of 2 of 3 fission product barriers with potential loss of third fission product barrier (eg, loss of Primary Coolant integrity, clad failure, and high potential for loss of Containment).	Current scheme would classify at lower threshold without regard to plant status. New scheme takes into account plant at cold shutdown and it takes longer for events to affect the public.
CA4.2.	With CONTAINMENT CLOSURE established and PCS integrity not established or PCS inventory reduced an UNPLANNED event results in PCS temperature exceeding the Technical Specification cold shutdown temperature limit of 200 °F for GREATER THAN 20 minutes[1]. [1]Note: if a PCS heat removal system is in operation within this time frame and PCS temperature is being reduced then this EAL is not applicable.	FPB/FD	**GE** Loss of 2 of 3 fission product barriers with potential loss of third fission product barrier (eg, loss of Primary Coolant integrity, clad failure, and high potential for loss of Containment).	Current scheme would classify at lower threshold without regard to plant status. New scheme takes into account plant at cold shutdown and it takes longer for events to affect the public.
CA4.3.	An UNPLANNED event results in PCS temperature exceeding the Technical Specification cold shutdown temperature limit of 200°F for GREATER THAN 60 minutes[1] or results in an PCS pressure rise of GREATER THAN 10 psig. [1]Note: if a PCS heat removal system is in operation within this time frame and PCS temperature is being reduced then this EAL is not applicable.	FPB/FD	**GE** Loss of 2 of 3 fission product barriers with potential loss of third fission product barrier (eg, loss of Primary Coolant integrity, clad failure, and high potential for loss of Containment).	Current scheme would classify at lower threshold without regard to plant status. New scheme takes into account plant at cold shutdown and it takes longer for events to affect the public.
CS1	Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability	FPB/FD		One to one corresponding category to IC
CS1.1.	With CONTAINMENT CLOSURE not established: a. Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 616 ft. 6 in. el. **OR** b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by unexplained Containment Sump or Primary System Drain Tank level rise	FPB/FD	**GE** Loss of 2 of 3 fission product barriers with potential loss of third fission product barrier (eg, loss of Primary Coolant integrity, clad failure, and high potential for loss of Containment).	Current scheme would classify at lower threshold without regard to plant status. New scheme takes into account plant at cold shutdown and it takes longer for events to affect the public.

CS1.2.	With CONTAINMENT CLOSURE established: a. Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 614 ft. 0 in. el. **OR** b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by either: • Unexplained Containment Sump or Primary System Drain Tank level rise • Erratic Source Range Monitor Indication	FPB/FD	**GE** Loss of 2 of 3 fission product barriers with potential loss of third fission product barrier (eg, loss of Primary Coolant integrity, clad failure, and high potential for loss of Containment).	Current scheme would classify at lower threshold without regard to plant status. New scheme takes into account plant at cold shutdown and it takes longer for events to affect the public.
IC CS2	Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel.	FPB/FD		One to one corresponding category to IC
CS2.1.	With CONTAINMENT CLOSURE not established, a. Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 616 ft. 6 in. **OR** b. Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by one or more of the following: • Containment High Range Radiation Monitor reading GREATER THAN 10 R/hr • Erratic Source Range Monitor Indication	FPB/FD	**GE** Loss of 2 of 3 fission product barriers with potential loss of third fission product barrier (eg, loss of Primary Coolant integrity, clad failure, and high potential for loss of Containment).	Current scheme would classify at lower threshold without regard to plant status. New scheme takes into account plant at cold shutdown and it takes longer for events to affect the public.
CS2.2.	With CONTAINMENT CLOSURE established a. Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 614 ft. 0 in. el. **OR** b. Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by any of the following: • Containment High Range Radiation Monitor reading GREATER THAN 10 R/hr • Erratic Source Range Monitor Indication	FPB/FD	**GE** Loss of 2 of 3 fission product barriers with potential loss of third fission product barrier (eg, loss of Primary Coolant integrity, clad failure, and high potential for loss of Containment).	Current scheme would classify at lower threshold without regard to plant status. New scheme takes into account plant at cold shutdown and it takes longer for events to affect the public.

IC CG1	Loss of Reactor Vessel Inventory Affecting Fuel Cladding Integrity with Containment Challenged and Irradiated Fuel in the Reactor Vessel	FPB/FD		One to one corresponding category to IC
CG1.1	Reactor vessel level: a. LESS THAN 614 ft. 0 in. for GREATER THAN 30 minutes **OR** b. cannot be monitored with indication of core uncovery for GREATER THAN 30 minutes as evidenced by any of the following • Containment High Range Radiation Monitor reading GREATER THAN 10 R/hr • Erratic Source Range Monitor indication • Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump or Primary System Drain Tank level rise **AND** Indication of CONTAINMENT challenged as indicated by one or more of the following: • Containment hydrogen concentration GREATER THAN OR EQUAL TO 3% • Containment pressure above 55 psig • CONTAINMENT CLOSURE not established	FPB/FD	**GE** Loss of 2 of 3 fission product barriers with potential loss of third fission product barrier (eg, loss of Primary Coolant integrity, clad failure, and high potential for loss of Containment).	Current scheme would classify at lower threshold without regard to plant status. New scheme takes into account plant at cold shutdown and it takes longer for events to affect the public.

Hazards

EAL#	EAL	Category	Current EAL	Comments
IC HU1	Natural and Destructive Phenomena Affecting the PROTECTED AREA.	Natural Phenomenon		One to one corresponding category to IC
HU1.1	Earthquake felt in plant as indicated by: Report by Plant Personnel to on duty Control Room Personnel AND Fluctuation in plant tank levels	Natural Phenomenon	Any earthquake felt in-Plant.	Same
HU1.2	Report by plant personnel of tornado or high winds GREATER THAN 100 mph striking within PROTECTED AREA boundary.	Natural Phenomenon	Tornado onsite	Wind speed is based on plant design.
HU1.3	Vehicle crash into plant structures or systems within PROTECTED AREA boundary.	Hazards-General	**UE** Aircraft crash onsite or unusual aircraft activity over facility which could affect Plant operation. **UE** Near or onsite train derailment which could affect Plant operation	New EAL includes all vehicles.
HU1.4	Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.	Hazards-General	**ALERT** Known explosion damage to facility affecting plant operation.	New EAL would classify event as UE instead of ALERT.
HU1.5	Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.	Secondary Side	**UE** Turbine rotating component failure **ALERT** Turbine failure causing casing penetration	New EAL would classify event as UE instead of ALERT.

HU1.6	Uncontrolled flooding in the following areas of the plant that has the potential to affect safety related equipment needed for the current operating mode: • Emergency Diesel Generator Rooms • Engineered Safeguards Rooms • Auxiliary Feedwater Pump Room • Switchgear Room 1C • Screen House • Component Cooling Water Pump Room	Natural Phenomenon	**UE** Abnormal water levels including flood or low water or seiche.	Similar classifications.
HU1.7	High lake level: ultimate heat sink level GREATER THAN 590 ft el. affecting the PROTECTED AREA. **OR** Low lake level: ultimate heat sink level LESS THAN OR EQUAL TO 572 ft 0 in. el. affecting the PROTECTED AREA.	Natural Phenomenon	**UE** Abnormal water levels including flood or low water or seiche.	Similar classifications.
IC HU2	FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection.	Fire		One to one corresponding category to IC
HU2.1	FIRE in buildings or areas contiguous to any of the following Table H-1 areas not extinguished within 15 minutes of control room notification or verification of a control room alarm: Table H-1 Plant VITAL Areas • Containment Structure • Auxiliary Building • Turbine Building • Screenhouse	Fire	Fire within the Plant lasting more than 10 minutes.	Current system uses 10 minutes instead of 15 and the building are defined instead of "within the Plant". Some events we would currently classify would be non-events.
IC HU3	Release of Toxic or Flammable Gases Deemed Detrimental to Normal Operation of the Plant.	Hazards-General None		New EAL added under this IC.
HU3.1	Report or detection of toxic or flammable gases that has or could enter the site area boundary in amounts that can affect NORMAL PLANT OPERATIONS.	Hazards-General	**UE** Near or onsite toxic or flammable gas which could affect Plant operation.	Similar classifications.
HU3.2	Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event.	None		No corresponding EAL in existing scheme

IC HU4	Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant.	Hazards-General Security		Two of the current categories apply to this IC.
HU4.1	Security Shift Leader reports ANY of the following: • Suspected sabotage device discovered within the plant Protected Area • Suspected sabotage device discovered outside the Protected Area or in the plant switchyard • Confirmed tampering with safety-related equipment • A hostage situation that disrupts normal plant operations • Civil disturbance or strike which disrupts normal plant operations • Internal disturbance that is <u>not</u> a short lived or that is not a harmless outburst involving ANY individuals within the Protected Area • Malevolent use of a vehicle outside the Protected Area which disrupts normal plant operations	Hazards-General Security	**UE** Aircraft crash onsite or unusual aircraft activity over facility which could affect Plant operation. **UE** Near or onsite train derailment which could affect Plant operation **UE** Near or onsite explosion which could affect Plant operation. **UE** Credible Security threat or attempted entry or attempted sabotage.	Similar classifications.
HU4.2	Credible Security Threat of "LOW" Severity	Security	**UE** Credible Security threat or attempted entry or attempted sabotage.	Similar classifications.
IC HU5	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a UE.	Miscellaneous		One to one corresponding category to IC
HU5.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.	Miscellaneous	**UE** Plant conditions exist that warrant increased awareness on the part of the Plant staff or state and/or local authorities, OR involve other than normal shutdown.	Similar classifications.
IC HA1	Natural and Destructive Phenomena Affecting the Plant VITAL AREA.	Natural Phenomenon Hazards-General		Two of the current categories apply to this IC.
HA1.1	Seismic event GREATER THAN 0.1 g Operating Basis Earthquake (OBE) as indicated by seismic instrumentation analysis	Natural Phenomenon	**ALERT** Any earthquake that exceeds operating base earthquake surface acceleration levels of 0.1G, but not greater than 0.2G; no damage to equipment required for safe shutdown.	

			SAE Any earthquake that exceeds safe shutdown earthquake surface acceleration levels or 0.2G.	
HA1.2	Tornado or high winds GREATER THAN 100 mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures / equipment (Table H-2) or Control Room indication of degraded performance of those systems. **Table H-2 VITAL AREAS/Safe Shutdown Equipment** • Containment Structure • Auxiliary Building • Turbine Building • Screenhouse • Safety Injection Refueling Water Tank • Condensate Storage Tank	Natural Phenomenon	**ALERT** Tornado striking facility. **SAE** Tornado or sustained winds in excess of design level; ie, of sufficient magnitude to cause damage to equipment and structures needed to safely shut down the Plant.	
HA1.3	Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures or equipment therein or control indication of degraded performance of those systems: (Table H-2)	Hazards-General	**SAE** Aircraft crash affecting Vital structures by impact or fire AND Plant not in cold shutdown.	New EALs do not have the cold shutdown criteria and would classify the event as an ALERT instead of the SAE unless safety systems are affected.
HA1.4	Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any of the following plant areas: (Table H-2)	Hazards-General	**SAE** Severe damage to equipment required for safe shutdown from missiles or explosion	Current EAL was intended to cover turbine-generated missiles. What was a SAE would be an ALERT unless safety systems are affected.

HA1.5	Uncontrolled flooding in following areas of the plant that results in degraded safety system performance as indicated in the control room or that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment: • Emergency Diesel Generator Rooms • Engineered Safeguards Rooms • Auxiliary Feedwater Pump Room • Switchgear Room 1C • Screen House • Component Cooling Water Pump Room	Natural Phenomenon	**ALERT** Flood, low water, or seiche near design basis. **SAE** Flood, low water, or seiche greater than design levels OR failure of protection of Vital equipment at lower levels.	
HA1.6	The following occurrences within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to plant structures containing equipment necessary for safe shutdown, or has caused damage as evidenced by control room indication of degraded performance of those systems. High lake level: ultimate heat sink level above 594 ft el. **OR** Low lake level: ultimate heat sink level lowers to 569 ft el.	Natural Phenomenon	**ALERT** Flood, low water, or seiche near design basis. **SAE** Flood, low water, or seiche greater than design levels OR failure of protection of Vital equipment at lower levels.	
HA2	FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown.	Fire		One to one corresponding category to IC.
HA2.1	FIRE or EXPLOSION in any of the following areas (Table H-2): **Table H-2 VITAL AREAS/ Safe Shutdown Equipment** • Containment Structure • Auxiliary Building • Turbine Building • Screenhouse • Safety Injection Refueling Water Tank • Condensate Storage Tank AND Affected system parameter indications show degraded performance or plant personnel report VISIBLE DAMAGE to permanent structures or equipment within the specified area.	Fire	**ALERT** Fire potentially affecting safety systems **SAE** Fire compromising the function of safety systems.	Some events we would currently classify as "potentially affecting safety system" could (if the fire is extinguished in less than 15 minutes) could be a non-event. Current scheme could classify events as a SAE which the new scheme would class an ALERT.
IC HA3	Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown.	Hazards/ General		One to one corresponding category to IC
HA3.1	Report or detection of toxic gases within or contiguous to a Safe Shutdown/VITAL AREA (Table H-2) in concentrations that may	Hazards/ General	**ALERT** Entry into facility environs of uncontrolled toxic	Some events we would currently classify as ALERT

	result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH). **Table H-2 VITAL AREAS/ Safe Shutdown Equipment** • Containment Structure • Auxiliary Building • Turbine Building • Screenhouse • Safety Injection Refueling Water Tank • Condensate Storage Tank		or flammable gas. **SAE** Entry of uncontrolled flammable gas into Vital areas OR entry of uncontrolled toxic gas into Vital areas that constitute a safety problem	could be UEs or non-events. Some events we would currently classify as SAEs would be ALERTs, UEs or non-events.
HA3.2	Report or detection of gases in concentration greater than the LOWER FLAMMABILITY LIMIT within or contiguous to a Safe Shutdown/VITAL AREA (Table H-2).	Hazards/ General	See above	See above
IC HA4	Confirmed Security Event in a Plant PROTECTED AREA.	Security		One to one corresponding category to IC
HA4.1	INTRUSION into the plant PROTECTED AREA by a HOSTILE FORCE.	Security	**ALERT** Security threat exists that results in adversaries commandeering an area of the Plant, but not control over shutdown capability or of any Vital areas.	Similar classifications.

HA4.2	Credible Security Threat of "HIGH" Severity OR Security Shift Leader reports any of the following: • Sabotage device discovered in the plant Protected Area • Standoff attack on the protected area by a hostile force (i.e., Sniper) • ANY Security event of increasing severity that persists for GREATER THAN 30 min.: o Credible bomb threats o Extortion o Suspicious Fire or Explosion o Significant Security System Hardware Failure o Loss of Guard Post Contact	Security	See ABOVE	Similar classifications.
IC HA5	Control Room Evacuation Has Been Initiated.	Evacuation		One to one corresponding category to IC
HA5.1	Entry into ONP-25.2, "Alternate Safe Shutdown" for control room evacuation.	Evacuation	**ALERT** Evacuation of Control Room anticipated or required with control of shutdown systems established at local stations	No change
IC HA6	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert.	Miscellaneous		One to one corresponding category to IC
HA6.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	Miscellaneous	**ALERT** Plant conditions exist that warrant precautionary activation of Technical Support Center and placing Emergency Operations Facility and other emergency personnel on standby.	Similar classifications.
IC HS1	Confirmed Security Event in a Plant VITAL AREA.	Security		One to one corresponding category to IC
HS1.1	INTRUSION into the plant VITAL AREA by a HOSTILE FORCE.	Security	**SAE** Physical attack on the Plant involving imminent occupancy of the Control Room, auxiliary shutdown panels, or other Vital areas.	Similar classifications.

HS1.2	Shift Security Leader reports confirmed sabotage discovered in a VITAL AREA.	None		No corresponding EAL in existing scheme
IC HS2	Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established.	Evacuation		One to one corresponding category to IC
HS2.1	Control room evacuation has been initiated. **AND** Control of the plant cannot be established per ONP 25.2, Alternate Safe Shutdown within 15 minutes.	Evacuation	**SAE** Evacuation of Control Room and control of shutdown systems not established at local stations within 15 minutes.	No change
IC HS3	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency.	Miscellaneous		One to one corresponding category to IC
HS3.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	Miscellaneous	**SAE** Plant conditions warrant the activation of State and County Emergency Operations Centers and monitoring teams or a precautionary notification to the public near the site.	Similar classifications
IC HG1	Security Event Resulting in Loss Of Physical Control of the Facility.	Security		One to one corresponding category to IC
HG1.1	A HOSTILE FORCE has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions as indicated by loss of physical control of EITHER: A VITAL AREA such that operation of equipment required for safe shutdown is lost OR Spent fuel pool cooling systems if imminent fuel damage is likely (e.g., freshly off-loaded reactor core in the pool).	Security	**GE** Physical attack on the Plant has resulted in unauthorized personnel occupying the Control Room or any other Vital areas.	Similar classifications.

IC HG2	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency.	Miscellaneous		One to one corresponding category to IC
HG2.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	Miscellaneous	**GE** Conditions exist that make release of large amounts of radioactivity in a short time possible (eg, any core melt situation).	Similar classifications

ISFSI

EAL#	EAL	Category	Current EAL	Comments
IC EU1	Damage to loaded cask CONFINEMENT BOUNDARY	Dry Fuel Storage	ISFSI=Independent Spent Fuel Storage Installation	One to one corresponding category to IC
EU1.1	Natural phenomena events affecting a loaded cask CONFINEMENT BOUNDARY as indicated by VISIBLE DAMAGE: • earthquake • tornado • flood • seiche • lightning	Dry Fuel Storage	**UE** Incident involving a loaded fuel storage cask - **OUTSIDE** the Auxiliary Building. Radiation level = 1 rem/hr at 1 ft from a Dry Fuel Storage Cask.	Old EAL was based on results of event and new is based on the event. There is potential for events to be classified as UEs under the new scheme which would have not been been classified under the current.
EU1.2	Accident conditions affecting a loaded cask CONFINEMENT BOUNDARY as indicated by VISIBLE DAMAGE: • tipped over or dropped cask • cask burial • fire • explosion	Dry Fuel Storage	**UE** Incident involving a loaded fuel storage cask - SED opinion based on direct observation that containment/shielding of a Dry Fuel Storage Cask has been degraded due to an operational event (cask drop, missile impact, etc).	Only one EAL under the currents scheme. The new one is more descriptive.
EU1.3	Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask CONFINEMENT BOUNDARY.	Dry Fuel Storage	See above	Similar to the current scheme.
IC EU2	Confirmed Security Event with potential loss of level of safety of the ISFSI.	Security	There is no separate group for security under the new EAL scheme.	One to one corresponding category to IC
EU2.1	Security Event as determined from PNP Security And Safeguards Contingency Plan and reported by the Security Shift Leader.	Security	**UE** Credible Security threat or attempted entry or attempted sabotage. Security alarms OR observation AND activation of Safeguards Contingency Procedures.	Specific security threat to Dry Fuel Storage in new scheme was covered in Security category in old scheme.

System Malfunction

EAL#	EAL	Category	Current EAL	Comments
IC SU1	Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes	Plant Power - Electrical		One to one corresponding category to IC
SU1.1.	Loss of all offsite power to both Vital 2400 VAC busses 1C and 1D for GREATER THAN 15 minutes. **AND** At least 1 emergency diesel generator is supplying power to each Vital 2400 VAC bus.	Plant Power - Electrical	**UE** Loss of offsite AC power. **UE** Loss of emergency onsite AC power. **ALERT** Loss of offsite AND onsite AC power for less than 15 minutes.	New EAL is more detailed, but what was an Alert would be an UE. If both diesels or offsite power is available, no classification would be made under the new scheme.
IC SU2	Inability to Reach Required Shutdown Within Technical Specification Limits	Engineered Safety Features		One to one corresponding category to IC
SU2.1.	Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time	Engineered Safety Features	**UE** Inability to reach a required mode within Technical Specification limits.	New EAL scheme contains more descriptive language.
IC SU3	UNPLANNED Loss of Most or All Safety System Annunciation or Indication in the Control Room for GREATER THAN 15 Minutes	Alarms/ Annunciators		One to one corresponding category to IC
SU3.1	UNPLANNED loss of most or all annunciators or indicators associated with safety systems for GREATER THAN 15 minutes on the following: • EC 02/12 – Reactor/PCPs • EC-03/13 – Safety Injection/Containment Cooling & Isolation systems / Ventilation • EC-04 – Electrical • EC-06 – RPS • EC-08 – Service Water/Component Cooling • EC-11 (Rear) – Rad Monitors • EC-11A (Front) – Control Room HVAC / Reactor Vessel Level / Core Exit Thermocouples	Alarms/ Annunciators	**ALERT** Loss of most or all alarms (annunciators) in Control Room.	New EAL is more detailed, but what was an Alert would be an UE.

IC SU4	Fuel Clad Degradation.	FPB/FD		One to one corresponding category to IC
SU4.1	Coolant sample activity GREATER THAN 40 µCi/gm dose equivalent I-131indicating fuel clad degradation	FPB/FD	**UE** Primary Coolant Iodine-131 dose concentration equivalent > 1.0µCi/gm for more than 72 hrs **ALERT** Primary Coolant Iodine-131 dose concentration equivalent >32µCi/gm OR > 5% total failed fuel.	Lower threshold for new EALs correspond to plant analysis and risk to the public.
IC SU5	PCS Leakage.	PCS Integrity		One to one corresponding category to IC
SU5.1	Unidentified or pressure boundary leakage GREATER THAN 10 gpm.	PCS Integrity	**UE** PCS leak rate in excess of Technical Specifications, but < 50 gpm. **Or** Primary to secondary leakage rate exceeds Technical Specifications limits of 432 gallons per day primary to secondary LEAKAGE through any one SG but < 50 gpm.	Higher limit is based on risk to the public and plant analysis.
SU5.2	Identified leakage GREATER THAN 25 gpm	PCS Integrity	See Above	Differentiates btw. Identified and unidentified leakage.
IC SU6	UNPLANNED Loss of All Onsite or Offsite Communications Capabilities	Comm. Loss		One to one corresponding category to IC
SU6.1.	Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations	Comm. Loss	**UE** Significant loss of offsite communication capability. Loss of the Emergency Notification System (ENS), and all other phones including satellite phones, that could be used to make notifications to Van Buren County, the State of Michigan and the NRC.	Only one EAL in the current scheme. We could classify some events under the new scheme as UEs that would not have been classified at all under the current scheme

SU6.2.	Loss of all Table C-2 offsite communications capability affecting the ability to perform routine operations	Comm. Loss	See above	See above
IC SU8	Inadvertent Criticality.	Engineered Safety Features		One to one corresponding category to IC
SU8.1	An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.	None		No corresponding EAL in existing scheme.
IC SA2	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 Successful	Engineered Safety Features		One to one corresponding category to IC
SA2.1.	Indication(s) exist that a Reactor Protection System setpoint was exceeded **AND** RPS automatic trip did **NOT** occur **AND** A successful manual trip occurred from: • EC-02 Reactor Trip pushbutton **OR** • EC-06 Reactor Trip pushbutton **OR** • Reactor tripped from an ATWS trip signal	Engineered Safety Features	**ALERT** Failure of the Reactor Protection System to initiate and complete a trip AND the Reactor is not sub critical.	Similar EALs.
IC 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.	Alarms/ Annunciators		One to one corresponding category to IC
SA4.1.	UNPLANNED loss of most or all annunciators or indicators associated with safety systems for GREATER THAN 15 minutes on the following: • EC 02/12 – Reactor/PCPs • EC-03/13 – Safety Injection/Containment Cooling & Isolation systems / Ventilation • EC-04 – Electrical • EC-06 – RPS • EC-08 – Service Water/Component Cooling • EC-11 (Rear) – Rad Monitors • EC-11A (Front) – Control Room HVAC / Reactor Vessel Level / Core Exit Thermocouples **AND** Either of the following: (a or b) a. A SIGNIFICANT TRANSIENT is in progress. **OR** b. Compensatory non-alarming indications are unavailable.	Alarms/ Annunciators	**SAE** Loss of most or all alarms (annunciators) in Control Room AND Plant transient initiated or in progress..	New EAL is more detailed, but what was a SAE would be an Alert.

IC 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	None		No corresponding EAL in existing scheme.
SA5.1.	AC power capability to VITAL 2400 VAC busses 1C and 1D reduced to only one of the following sources for GREATER THAN 15 minutes • Safeguard Transformer 1-1 • Start Up Transformer 1-2 • 1-1 Emergency Diesel Generator • 1-2 Emergency Diesel Generator **AND** Any additional single failure will result in station blackout.	None		No corresponding EAL in existing scheme. Current EAL would classify event as a SAE.
IC SS1	Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses	Plant Power - Electrical		One to one corresponding category to IC
SS1.1.	Loss of all offsite power to VITAL 2400 VAC buses 1C and 1D. **AND** Failure of all emergency generators to supply power to VITAL 2400 VAC buses. **AND** Failure to restore power to at least one VITAL 2400 VAC bus within 15 minutes from the time of loss of both offsite and onsite AC power.	Plant Power - Electrical	**SAE** Loss of offsite AND onsite AC power for more than 15 minutes	Similar EALs.
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	Engineered Safety Features		One to one corresponding category to IC
SS2.1.	Indication(s) exist that automatic and manual trip were NOT successful in reducing power to LESS THAN 2.5%.	Engineered Safety Features	**SAE** Transient requiring operation of shutdown systems with failure to trip (continued power generation, but no core damage immediately evident).	Similar EALs. New EAL adds the "2.5%" power level.

IC SS3	Loss of All Vital DC Power	Plant Power - Electrical		One to one corresponding category to IC
SS3.1.	Loss of all vital DC power based on LESS THAN 105 VDC on DC buses No. 1, ED-10 and No.2, ED-20 for GREATER THAN 15 minutes.	Plant Power - Electrical	**ALERT** Loss of all onsite DC power for less than 15 minutes. **SAE** Loss of all vital onsite DC power for more than 15 minutes, AND Plant not in cold shutdown.	ALERT classification is not included.
IC SS4	Complete Loss of Heat Removal Capability	Engineered Safety Features		One to one corresponding category to IC
SS4.1.	Loss of core cooling and heat sink.	Engineered Safety Features	Complete loss of any function needed for hot shutdown.	New EAL is more descriptive.
IC SS6	Inability to Monitor a SIGNIFICANT TRANSIENT in Progress.	Alarms/ Annunciators		One to one corresponding category to IC
SS6.1.	Loss of most or all annunciators associated with safety systems on the following: • EC 02/12 – Reactor/PCPs • EC-03/13 – Safety Injection/Containment Cooling & Isolation systems / Ventilation • EC-04 – Electrical • EC-06 – RPS • EC-08 – Service Water/Component Cooling • EC-11 (Rear) – Rad Monitors • EC-11A (Front) – Control Room HVAC / Reactor Vessel Level / Core Exit Thermocouples **AND** SIGNIFICANT TRANSIENT in progress. **AND** Compensatory non-alarming indications are unavailable. **AND** Indications needed to monitor the ability to shut down the reactor, maintain the core cooled, maintain the primary coolant system intact, and maintain containment intact are unavailable.	Alarms/ Annunciators	**SAE** Loss of most or all alarms (annunciators) in Control Room AND Plant transient initiated or in progress.	Current EAL would have declared a SAE before these conditions were met.

IC SG1	Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses.	Plant Power - Electrical		One to one corresponding category to IC
SG1.1.	Loss of all offsite power to Vital 2400 VAC buses 1C and 1D. **AND** Failure of both emergency diesel generators to supply power to Vital 2400 VAC buses. **AND** Either of the following: (a or b) a. Restoration of at least one Vital 2400 VAC bus within 4 hours is <u>not</u> likely **OR** b. Continuing degradation of core cooling based on Fission Product Barrier monitoring as indicated by Average of qualified CETs GREATER THAN 700 °F.	Plant Power - Electrical	**GE** Loss of offsite and onsite AC power AND total loss of auxiliary feedwater makeup capability for greater than 2 hours.	Classifies at SAE. If FPBs are threatened would go to FPB Table.
IC SG2	Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core	Engineered Safety Features		One to one corresponding category to IC
SG2.1.	Indication(s) exist that automatic and manual trip were NOT successful in reducing power to LESS THAN 2.5%. **AND** Either of the following: (a or b) a. Indication(s) exists that the core cooling is extremely challenged as indicated by Average of qualified CETs GREATER THAN 1200 °F. **OR** b. Indication(s) exists that heat removal is extremely challenged as indicated by Core and PCS heat removal safety function status acceptance criteria <u>cannot</u> be met.	Engineered Safety Features	**GE** Transient initiated by loss of Feedwater and Condensate System (principal heat removal system), followed by failure of Auxiliary Feedwater System for an extended period. Core melting possible in several hours.	New EAL give more descriptive terms for classification.

ENCLOSURE 4

DETAILED SUPPORTING INFORMATION

ATTACHMENT 1

RED-LINE TECHNICAL BASIS DOCUMENT

164 Pages Follow

Table of Contents

ACRONYMS

AC	Alternating Current
ATWS	Anticipated Transient Without Scram
CCW	Component Cooling Water
CDE	Committed Dose Equivalent
CE	Combustion Engineering
CFR	Code of Federal Regulations
CMT	Containment
CSF	Critical Safety Function
DC	Direct Current
DHR	Decay Heat Removal
DOT	Department of Transportation
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
ECL	Emergency Classification Level
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
EI	Emergency Implementing Procedure
EPRI	Electric Power Research Institute
ESF	Engineered Safeguards Feature
FEMA	Federal Emergency Management Agency
FSAR	Final Safety Analysis Report
GE	General Emergency
HPSI	High Pressure Safety Injection
IC	Initiating Condition
IDLH	Immediately Dangerous to Life and Health
IPEEE	Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI	Independent Spent Fuel Storage Installation
LCO	Limiting Condition of Operation
LER	Licensee Event Report
LFL	Lower Flammability Limit
LOCA	Loss of Coolant Accident

LPSI	Low Pressure Safety Injection
MSIV	Main Steam Isolation Valve
mRem	milliRem
Mw	Megawatt
NEI	Nuclear Energy Institute
NESP	National Environmental Studies Project
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Management and Resources Council
OBE	Operating Basis Earthquake
ODCM	Offsite Dose Calculation Manual
PCS	Primary Coolant System
PNP	Palisades Nuclear Plant
PPC	Palisades Plant Computer
PRA/PSA	Probabilistic Risk Assessment / Probabilistic Safety Assessment
PSIG	Pounds per Square Inch Gauge
PWR	Pressurized Water Reactor
R	Rem
RG	Regulatory Guide
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RVLMS	Reactor Vessel Level Monitoring System
SAE	Site Area Emergency
SG	Steam Generator
SI	Safety Injection
SRO	Senior Reactor Operator
SSE	Safe Shutdown Earthquake
SWS	Service Water System
TEDE	Total Effective Dose Equivalent
TOAF	Top of Active Fuel
TSC	Technical Support Center
UE	Notification Of Unusual Event

1.0 PURPOSE

This document provides the detailed set of Emergency Action Levels (EALs) applicable to the Palisades Nuclear Plant (PNP) and the associated Technical Bases using the EAL development methodology found in NEI 99-01 Revision 4 [Ref. 2.1].

The primary tool for determining the emergency classification level is the Emergency Action Level Matrix. Personnel responsible for implementation of EI-1, Emergency Classification and Actions [Ref. 2.2], and the EAL Matrix [Ref. 2.3] may use this document as a technical reference and an aid in EAL interpretation.

The user of the EAL Matrix may (but is not required) to consult the EAL Technical Basis Document in order to obtain additional information concerning the EALs under classification consideration.

2.0 REFERENCES

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

2.2 EI-1, "Emergency Classification and Actions"

2.3 Emergency Action Level Matrix

2.4 PNP Technical Specifications Table 1.1-1

2.5 GOP-14, "Shutdown Cooling Operations" Attachment 1 - Terms and Definitions.

2.6 FSAR Figure 1-1, Plant Area Plan

2.7 PNP Site Emergency Plan

3.0 DISCUSSION

3.1 Background

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the PNP Emergency Plan [Ref. 2.7].

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

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

- Consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur during plant shutdown conditions.

- Addressing initiating conditions and example emergency action levels that fully address conditions that may be postulated to occur at permanently Defueled Stations and Independent Spent Fuel Storage Installations.

- Simplifying the fission product barrier EAL threshold for a Site Area Emergency.

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

3.2 Key Definitions in EAL Methodology

The following definitions apply to the generic EAL methodology:

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

- Notification of Unusual Event (UE)

- Alert

- Site Area Emergency (SAE)

- General Emergency (GE)

Section 3.4 provides further discussion of the emergency classes.

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

- An IC is an emergency condition, which sets it apart from the broad class of conditions that may or may not have the potential to escalate into a radiological emergency.

- It can be a continuous, measurable function that is outside technical specifications, such as elevated PCS temperature or falling reactor coolant level (a symptom).

- It also encompasses occurrences such as FIRE (an event) or reactor coolant pipe failure (an event or a barrier breach).

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

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

3.3 Recognition Categories

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

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

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

3.4 Emergency Class Descriptions

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

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

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

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

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

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

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

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

SITE AREA EMERGENCY: Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels, which exceed EPA PAG exposure levels beyond the site boundary.

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

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

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

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

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

3.5 Operating Mode Applicability

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

1 Power Operations

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

2 Startup

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

3 Hot Standby

K_{eff} LESS THAN 0.99 average primary coolant temperature (Tave) GREATER THAN OR EQUAL TO 300°F

4 Hot Shutdown

K_{eff} LESS THAN 0.99 and average primary coolant temperature (Tave) LESS THAN 300°F and GREATER THAN 200°F with all reactor vessel head closure bolts fully tensioned

5 Cold Shutdown

K_{eff} LESS THAN 0.99 and average primary coolant temperature (Tave) LESS THAN OR EQUAL TO 200°F with all reactor vessel head closure bolts fully tensioned

6 Refuel

One or more reactor vessel head closure bolts less than fully tensioned

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

D Defueled

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

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

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

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

3.6 Fission Product Barriers

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

The primary fission product barriers are:

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

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

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

3.7 Emergency Classification Based on Fission Product Barrier Degradation

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

- Notification of Unusual Event:

  Any loss or any potential loss of Containment

- Alert:

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

- Site Area Emergency:

  Loss or potential loss of any two barriers

- General Emergency:

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

3.8 EAL Relationship to Emergency Operating Procedures (EOPs)

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

3.9 Symptom Based vs. Event Based Approach

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

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

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

3.10 Treatment of Emergency Class Upgrading

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

3.11 Emergency Class Downgrading

Another important aspect of usable EAL guidance is the consideration of what to do when the risk posed by an emergency is clearly decreasing.

It is recommended that a combination approach be taken involving recovery from General Emergencies and some Site Area Emergencies and termination from UEs, Alerts, and certain Site Area Emergencies causing no long-term plant damage. Downgrading to lower emergency classes adds notifications but may have merit under certain circumstances.

3.12 Classifying Transient Events

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

In general, observe the following guidance: Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met. For example, a momentary event, such as an ATWS or an earthquake, requires declaration even though the condition may have been resolved by the time the declaration is made.

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

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

3.13 Imminent EAL Thresholds

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

4.0 TECHNICAL BASES INFORMATION

4.1 Recognition Category Organization

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

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

4.2 Initiating Condition Structure

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

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

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

- IC Description

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

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

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

4.3 EAL Identification

The EAL identifier is the IC identifier followed by a period and sequence number (e.g., RU1.1, RU1.2, etc.).

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

5.0 DEFINITIONS

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

AFFECTING SAFE SHUTDOWN: event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable HOT or COLD SHUTDOWN condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

> Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in HOT SHUTDOWN. HOT SHUTDOWN is achievable, but COLD SHUTDOWN is not. This event is not "AFFECTING SAFE SHUTDOWN."
>
> Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in COLD SHUTDOWN. HOT SHUTDOWN is achievable, but COLD SHUTDOWN is not. This event is "AFFECTING SAFE SHUTDOWN."

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

CIVIL DISTURBANCE/STRIKE: A CIVIL DISTURBANCE may involve a group of unexpected or unauthorized individuals outside of the OWNER CONTROLLED AREA or PROTECTED AREA or a planned gathering as a part of a labor dispute (e.g., labor picket line, protest demonstration, etc.). When such disturbance poses a potential threat to plant safety or personnel, additional protective measures shall be reviewed and, where appropriate, implemented.

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

CONTAINMENT CLOSURE is a containment condition where at least one integral barrier to the release of radioactive material is being provided. CONTAINMENT CLOSURE Controls are used to track any impaired containment penetration so that at least one barrier to the release of radioactive material can be quickly established in the event of a loss of decay heat removal. [Ref. 2.5]

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

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

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

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

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

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

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

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

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

NORMAL PLANT OPERATIONS: activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or

administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONS.

OWNER CONTROLLED AREA is the area surrounding the Plant in which the reactor licensee has the authority to determine all activities including exclusion or removal of persons and property from the area during accident conditions.

PROTECTED AREA boundary is within the security isolation zone and is defined in FSAR Figure 1-1, Plant Area Plan. [Ref. 2.6]

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

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

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

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

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

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

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

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

6.0 EMERGENCY ACTION LEVEL TECHNICAL BASIS

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

Recognition Category AR

Abnormal Rad Levels / Radiological Effluent

INITIATING CONDITION MATRIX

~~NOUE~~		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY	
RAU1	Any UNPLANNED Release of Gaseous or Liquid Radio-activity to the Environment that Exceeds Two Times the Offsite Dose Calculation Manual (ODCM) Limits ~~Radiological Effluent Technical Specifications~~ for 60 Minutes or Longer. *Op. Modes: All*	RAA1	Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Offsite Dose Calculation Manual (ODCM) Limits ~~Radiological Effluent Technical Specifications~~ for 15 Minutes or Longer. *Op. Modes: All*	RAS1	Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the Actual or Projected Duration of the Release. *Op. Modes: All*	RAG1	Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology. *Op. Modes: All*
RAU2	Unexpected Increase in Plant Radiation. *Op. Modes: All*	RAA3	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 *Op. Modes: All*				
		RAA2	Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel. *Op. Modes: All*				

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ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RAU1

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the ~~Radiological Effluent Technical Specifications~~Offsite Dose Calculation Manual (ODCM) Limits for 60 Minutes or Longer.

Operating Mode Applicability: All

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

RU1.1. VALID reading on ~~any~~ effluent monitor RIA-1049* "Liquid Radwaste Discharge Monitor" that exceeds two times the alarm setpoint* established by a current radioactivity discharge permit for 60 minutes or longer.
* with waste discharge not isolated

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

Stack Normal Range Monitor (RIA-2326)	3.2E+5 cpm
S/G Blowdown Monitor (RIA-0707)	2 X High Alarm*
Service Water Monitor (RIA-0833)	2 X High Alarm*
Turbine Bldg. Sumps Monitor (RIA-5211)	2 X High Alarm*

* with waste discharge not isolated

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

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

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

Basis:

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

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

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

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

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

~~EAL #2~~RU1.2 is also intended for ~~licensees that have established~~ effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared. The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. These monitor reading EALs ~~should be~~have been determined using this methodology. The values given are two times the ODCM release limits. [Ref. 1, 3]

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

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

~~EALs #1 and #2~~RU1.1 and RU1.2 directly correlates with the IC since annual average meteorology is required to be used in showing compliance with the ~~RETS~~ ODCM and is used in calculating the alarm setpoints. ~~EALs #4 and #5 are a function of actual meteorology, which will likely be different from the limiting annual average value. Thus, there will likely be a numerical inconsistency. However, the~~The fundamental basis of this IC is NOT a dose or dose rate, but rather the degradation in the level of safety of the plant implied by the uncontrolled release.
~~Exceeding EAL #4 or EAL #5 is an indication of an uncontrolled release meeting the fundamental basis for this IC.~~

PNP Basis Reference(s):

1. NMC calculation EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4"
2. PNP ODCM
3. RGC 85-003, Setpoint Basis Correspondence RG Christie, June 28, 1985

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

ARU2

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Unexpected Rise ~~Increase~~ in Plant Radiation.

Operating Mode Applicability: All

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

RU2.1. ~~a.~~ VALID ~~(site-specific)~~ indication of uncontrolled water level ~~decrease~~ lowering to LESS THAN 646 ft. elevation in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water.

AND

~~b. Unplanned~~ UNPLANNED VALID ~~(site-specific)~~ ~~Direct~~ Area Radiation Monitor reading ~~increases~~ rises ~~as indicated by alarm~~ on any of the following:

- Vent Monitor Fuel Handling Area (RIA-5712)
- Spent Fuel Pool Area Radiation Monitors (RIA-5709 or RIA-2313)
- Refueling Containment High Radiation (CHR) Monitors (RIA-2316 or RIA-2317)

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

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

Basis:

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

In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR, explicit coverage of these types of events via ~~EAL #1~~RU2.1 is appropriate given their potential for increased doses to plant staff. Classification as an ~~NO~~UE is warranted as a precursor to a more serious event. ~~Site-specific i~~Indications ~~may~~ include instrumentation such as water level and local area radiation monitors, and personnel (e.g., refueling crew) reports. Spent Fuel Pool Area Radiation Monitors (RIA-5709 and RIA-2313) and Refueling Containment High Radiation (CHR) Monitors (RIA-2316 or RIA-2317) also cover

incidents involving the fuel transfer canal. [Ref. 4, 5] If available, ~~security~~ video cameras may allow remote observation. Depending on available level instrumentation, the declaration threshold may need to be based on indications of water makeup rate or decrease in refueling water storage tank level.

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

The minimum allowable water level in the Spent Fuel Pool (SFP) and Refueling Cavity threshold is the ~~The RU2.1 water level threshold is the Technical Specification minimum allowable water level in the Spent Fuel Pool and Refueling Cavity.~~ Low Spent Fuel Pool water level, ~~is~~ alarmed in the Control Room (annunciator EK-1309) at 646 ft elevation or 35 ft above the bottom of the pool. [Ref. 1, 2, 3]

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

PNP Basis Reference(s):

1. Technical Specifications 3.7.14 Spent Fuel Pool (SFP) Water Level
2. Technical Specifications 3.9.6 Refueling Cavity Water Level
3. ARP-8 Annunciator #9 Spent Fuel Pool Hi/Lo Level
4. FSAR Table 11-16, "Area Radiation Detectors"
5. FSAR Table 11-15, "Process Radiation Service And Equipment"

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

ARA1

Initiating Condition -- ALERT

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the ~~Radiological Effluent Technical Specifications~~Offsite Dose Calculation Manual (ODCM) Limits for 15 Minutes or Longer.

Operating Mode Applicability: All

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

RA1.1. VALID reading on ~~any~~ effluent monitor RIA-1049 "Liquid Radwaste Discharge Monitor" that exceeds 200 times the alarm setpoint* established by a current radioactivity discharge permit for 15 minutes or longer.
* with waste discharge not isolated

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

Stack Normal Range Monitor (RIA-2326)	1.3E+6 cpm
Stack High Range Monitor (RIA-2327)	1.5 mRem/hr
S/G Blowdown Monitor (RIA-0707)	200 X High Alarm*
Service Water Monitor (RIA-0833)	200 X High Alarm*
Turbine Bldg. Sumps Monitor (RIA-5211)	200 X High Alarm*

* with waste discharge not isolated

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

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

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

Basis:

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

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

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

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

~~EAL #1~~RA1.1 addresses radioactivity releases that for whatever reason cause effluent radiation monitor readings that exceed two hundred times the alarm setpoint established by the radioactivity discharge permit. ~~This~~ The alarm setpoints may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance with the ~~RETS~~ODCM. Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

~~EAL #2 is similar to EAL #1, but is~~RA1.2 ~~is also~~ intended to address effluent or accident radiation monitors on non-routine release pathways (i.e., for which a discharge permit would not normally be prepared). The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. These monitor reading EALs ~~should be~~have been determined using this methodology [Ref. 1, 3]. The limit for RIA-2326 and RIA-2327 were selected to maintain the "intervals between EALs for the four classifications" [Ref. NEI 99-1 AS1 basis].

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

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

~~EALs #1 and #2~~RA1.1 and RA1.2 directly correlates with the IC since annual average meteorology is required to be used in showing compliance with the ~~RETS~~ ODCM and is used in calculating the alarm setpoints. ~~EALs #4 and #5 are a function of actual meteorology, which will likely be different from the limiting annual average value. Thus, there will likely be a numerical inconsistency. However, the~~The fundamental basis of this IC is NOT a dose or dose rate, but rather the degradation in the level of safety of the plant implied by the uncontrolled release. ~~Exceeding EAL #4 or EAL #5 is an indication of an uncontrolled release meeting the fundamental basis for this IC.~~

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

PNP Basis Reference(s):

1. NMC calculation EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4"
2. PNP ODCM
3. RGC 85-003, Setpoint Basis Correspondence RG Christie, June 28, 1985

Initiating Condition -- ALERT

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

Operating Mode Applicability: All

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

RA2.1. A VALID ~~(site-specific)~~ alarm or reading on ~~one or more~~ any of the following radiation monitors resulting from damage to irradiated fuel or loss of water level:

• Vent Monitor Fuel Handling Area RIA-5712	1E+4 cpm
• Spent Fuel Pool Area Radiation Monitors RIA-5709 or RIA-2313	15 mRem/hr
• Refueling Containment High Radiation (CHR) Monitors RIA-2316 or RIA-2317	80 mRem/hr above background

~~(site-specific monitors)~~ ~~Refuel Floor Area Radiation Monitor~~
~~Fuel Handling Building Ventilation Monitor~~
~~Refueling Bridge Area Radiation Monitor~~

RA2.2. Water level less than ~~(site-specific) feet~~ 636 ft. 9 in. elevation for the reactor refueling cavity, spent fuel pool and fuel transfer canal that will result in irradiated fuel uncovering.

Basis:

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

~~EAL #1~~RA2.1 addresses radiation monitor indications [Ref. 1, 2, 3, 4] of fuel uncovery and/or fuel damage. Increased readings on ventilation monitors may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background at the monitor due to water level decrease may mask increased ventilation exhaust airborne activity and needs to be considered. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the monitor could in fact be properly responding to a known event involving transfer or relocation of a source, stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Application of these Initiating Conditions requires understanding of the actual radiological conditions present in the vicinity of the monitor. Information Notice No. 90-08, *"KR-85*

Hazards from Decayed Fuel" ~~should be~~was considered in establishing radiation monitor EAL thresholds.

In ~~EAL #2~~RA2.2, ~~site-specific~~ indications ~~may~~ include instrumentation such as water level and local area radiation monitors, and personnel (e.g., refueling crew) reports. If available, ~~security~~ video cameras may allow remote observation. ~~Depending on available level indication, the d~~Declaration ~~threshold~~ may need to be based on indications of water makeup rate or decrease in refueling water storage tank level. [Ref. 5]

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

PNP Basis Reference(s):

1. ARP 8, Safeguards Safety Injection and Isolation Scheme EK 9 (EC 9), Annunciator No. 66
2. RI-86E-1, Refueling Isolation Monitors Calibration - Source Test
3. FSAR Table 11-16, "Area Radiation Detectors"
4. FSAR Table 11-15, "Process Radiation Service And Equipment"
5. EA-KFK-90-01 "Fuel Submergence vs. Fuel Handling Activities"

Initiating Condition -- ALERT

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

Operating Mode Applicability: All

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

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

Control Room (RIA-2310)

OR

Central Alarm Station (RIA-2304)

~~(Site-specific) list~~

RA3.2. Any VALID ~~(site-specific)~~ radiation monitor readings GREATER THAN ~~<site-specific> values312~~ R/hr in areas requiring infrequent access to maintain plant safety functions (Table H-1).

Table H-1 PLANT VITAL AREAS
• Containment Structure • Auxiliary Building • Turbine Building • Screenhouse

Basis:

This IC addresses increased radiation levels that impede necessary access to operating stations, or other areas containing equipment that must be operated manually or that requires local monitoring, in order to maintain safe operation or perform a safe shutdown. It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant. The cause and/or magnitude of the increase in radiation levels is not a concern of this IC. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other IC may be involved. For example, a dose rate of 15 mRem/hr in the control room may be a problem in itself. However, the increase may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, an SAE or GE may be indicated by the fission product barrier matrix ICs.

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

This IC is not meant to apply to increases in the containment ~~dome~~ radiation monitors as these are events which are addressed in the fission product barrier matrix ICs. Nor is it intended to apply to anticipated temporary increases due to planned events (e.g., incore detector movement, radwaste container movement, depleted resin transfers, etc.)

Areas requiring continuous occupancy includes the C~~c~~ontrol R~~r~~oom and ~~, as appropriate to the site,~~ any other control stations that are manned continuously, such as ~~a radwaste control room or a~~ the central security alarm station (CAS). CAS has no installed radiation monitoring capability however, RIA-2304 will provide indication of increasing radiation levels prompting surveys. [Ref. 1, 2] The value of 15mRem/hr is derived from the GDC 19 value of 5 R~~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 mRem/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an Alert.

For RA3.2 areas requiring infrequent access, the basis of the 3 R/hr value is as follows:

> The PNP annual administrative personnel exposure limit is 2 R/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 R without exceeding the legal 10CFR20 annual exposure limit of 5 R and thus the need for emergency exposure authorization. Assuming that an activity required to be performed in the plant would require a 60-minute stay time in that area an area exposure rate of 3 R/hr would not unduly impede access to areas necessary for safe plant shutdown.

~~the site-specific value(s) should be based on radiation levels which result in exposure control measures intended to maintain doses within normal occupational exposure guidelines and limits (i.e., 10 CFR 20), and in doing so, will impede necessary access.~~ As used here, *impede,* includes hindering or interfering provided that the interference or delay is sufficient to significantly threaten the safe operation of the plant. Table H-1 provides the list of safe shutdown areas requiring infrequent access. The listed areas contain functions and systems required for the safe shutdown of the plant. [Ref 3]

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

PNP Basis Reference(s):

1. ARP 8, "Safeguards Safety Injection and Isolation Scheme EK 13 (EC 13)"
2. SOP-39, "Area Radiation Monitoring System"
3. ONP-25.1, Fire Which Threatens Safety-Related Equipment

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

ARS1

Initiating Condition -- SITE AREA EMERGENCY

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

Operating Mode Applicability: All

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

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

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

Stack High Range Monitor (RIA-2327)	5.2 mRem/hr
Main Steam Line Monitors (RIA-2323/RIA-2324)	1.70E+3 cpm
Stack High Range Effluent Monitor (RIA-2328)*	0.64 mRem/hr
Atmospheric Dump Valve High Range Effluent Monitors (RIA-2328) *	0.57 mRem/hr

*Note RIA-2328 is a backup monitor which can be physically directed for monitoring the ADVs or Stack

~~(site-specific list)~~

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

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

RS1.34. Field survey results indicate closed window dose rates exceeding 100 mRem/hr expected to continue for more than one hour, at or beyond the site boundary;

~~or~~ OR

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

Basis:

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

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

The 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 ~~(site specific)~~ monitor list in ~~EAL #1~~ RS1.1 ~~should~~ includes monitors on all potential release pathways.

The EPA PAGs are expressed in terms of the sum of the *effective dose equivalent (EDE)* and the *committed effective dose equivalent (CEDE)*, or as the thyroid *committed dose equivalent (CDE).* For the purpose of these IC/EALs, the dose quantity *total effective dose equivalent (TEDE),* as defined in 10 CFR 20, is used in lieu of *"...sum of EDE and CEDE...."* The EPA PAG guidance provides for the use adult thyroid dose conversion factors.
~~However, some states have decided to calculate child thyroid CDE. Utility IC/EALs need to be consistent with those of the states involved in the facility's emergency planning zone.~~

The "SAE" effluent monitor readings are derived from Reference 1-NMC calculation, EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4." ~~The monitor reading EALs should be determined using a dose assessment method that back calculates from the dose values specified in the IC. The meteorology and source term (noble gases, particulates, and halogens) used should be the same as those used for determining the monitor reading EALs in ICs AU1 and AA1. This protocol will maintain intervals between the EALs for the four classifications. Since doses are generally not monitored in real time, it is suggested that a release duration of one hour be assumed, and that the EALs be based on a site boundary (or beyond) dose of 100 mR/hour whole body or 500 mR/hour thyroid, whichever is more limiting (as was done for EALs #3 and #4). If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.~~

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

PNP Basis Reference(s):

1. NMC calculation EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4"

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

ARG1

Initiating Condition -- GENERAL EMERGENCY

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

Operating Mode Applicability: All

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

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

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

Stack high range monitor (RIA-2327)	52 mRem/hr
Main Steam Line Monitors (RIA-2323/RIA-2324)	1.70E+4 cpm
Stack high range Effluent Monitor (RIA-2328)*	6.4 mRem/hr
Atmospheric Dump Valve High Range Effluent Monitors (RIA-2328)*	5.7 mRem/hr

*Note RIA-2328 is a backup monitor which can be physically directed for monitoring the ADVs or Stack

~~(site-specific list)~~

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

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

RG1.34. Field survey results indicate closed window dose rates exceeding 1000 mRem/hr expected to continue for more than one hour, at or beyond site boundary.

~~; or a~~

OR

Analyses of field survey samples indicate thyroid CDE of 5000 mRem for one hour of inhalation, at or beyond site boundary.

Basis:

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

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

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

The ~~(site-specific)~~ monitor list in ~~EAL #1~~RG1.1 ~~should~~ includes monitors on all potential release pathways.

The EPA PAGs are expressed in terms of the sum of the *effective dose equivalent (EDE)* and the *committed effective dose equivalent (CEDE)*, or as the thyroid *committed dose equivalent (CDE)*. For the purpose of these IC/EALs, the dose quantity *total effective dose equivalent (TEDE)*, as defined in 10 CFR 20, is used in lieu of *"...sum of EDE and CEDE...."* The EPA PAG guidance provides for the use adult thyroid dose conversion factors. ~~However, some states have decided to calculate child thyroid CDE. Utility IC/EALs need to be consistent with those of the states involved in the facilities emergency planning zone.~~

The ~~Table R-1 column~~ effluent monitor readings are derived from Reference 1 - NMC calculation EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01, Revision 4."
~~The monitor reading EALs should be determined using a dose assessment method that backcalculates from the dose values specified in the IC. The meteorology and source term (noble gases, particulates, and halogens) used should be the same as those used for determining the monitor reading EALs in ICs AU1 and AA1. This protocol will maintain intervals between the EALs for the four classifications. Since doses are generally not monitored in real-time, it is suggested that a release duration of one hour be assumed, and that the EALs be based on a site boundary (or beyond) dose of 1000 mR/hour whole body or 5000 mR/hour thyroid, whichever is more limiting (as was done for EALs #3 and #4). If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is released, the longer duration should be used.~~

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

PNP Basis Reference(s):

1. NMC calculation EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4"

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

INITIATING CONDITION MATRIX

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

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

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SYSTEM MALFUNCTION

CU1

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

~~RCS~~PCS Leakage.

Operating Mode Applicability: Cold Shutdown

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

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

CU1.2. Identified leakage GREATER THAN ~~greater than~~ 25 gpm.

Basis:

This IC is included as a ~~NO~~UE because it is considered to be a potential degradation of the level of safety of the plant.

The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is sufficiently large to be observable via normally installed instrumentation ~~(e.g., Pressurizer level, RCS loop level instrumentation, etc...)~~ or reduced inventory instrumentation such as level hose indication. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. Prolonged loss of ~~RCS~~PCS Inventory may result in escalation to the Alert level via either IC CA1 (Loss of ~~RCS~~PCS Inventory with Irradiated Fuel in the ~~RPV~~Reactor Vessel) or CA4 (Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the ~~RPV~~Reactor Vessel).

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

PNP Basis Reference(s):

None

SYSTEM MALFUNCTION

CU2

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

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

Operating Mode Applicability: Refueling

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

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

CU2.2. ~~a.~~ Loss of ~~RPV~~ Reactor Vessel inventory as indicated by unexplained Containment Sump OR Primary System Drain Tank ~~(site-specific) sump and tank~~ level ~~increase~~rise

AND

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

Basis:

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

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

In the refueling mode, normal means of core temperature indication and ~~RCS~~PCS level indication may not be available. Redundant means of ~~RPV~~Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of ~~RCS~~PCS inventory event, the operators would need to determine that ~~RPV~~Reactor Vessel inventory loss was occurring by observing ~~sump~~ Containment Sump or Primary System Drain Tank ~~and tank~~ level changes. ~~Sump~~ Sump and tank level ~~increase~~rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are

indicative of ~~RCS~~PCS leakage. Rising Containment Sump level can be monitored on the PPC [Ref. 1]. Rising Primary System Drain Tank level can be monitored at EC-40, "Radwaste Panel" annunciation for which will alarm both locally and in the Control Room. [Ref. 2, 3, 4] Escalation to Alert would be via either CA2 or ~~RCS~~PCS heatup via CA4.

~~EAL 1~~CU2.1 involves a decrease in ~~RCS~~PCS level below the top of the ~~RPV~~Reactor Vessel flange that continues for 15 minutes due to an UNPLANNED event. The Reactor Vessel flange is at 624 ft. 6 in. elevation and can be monitored by:

- Reactor Vessel Level indicator LIA-0105 wide range 65%
- Refueling Level Gauges LG-0105B 624 ft. 6 in. elevation. [Ref. 5]
- RVLMS UGS Region Sensor #5 uncovered (red light on): 102 in. above (fuel) bottom of fuel alignment plate or 621 ft. 8 in. elevation, which would provide additional indication that the EAL had been exceeded. [Ref. 6, 7]

This EAL is not applicable to decreases in flooded reactor cavity level (covered by ~~AU2 EAL1~~RU2.1) until such time as the level decreases to the level of the vessel flange. ~~For BWRs, if RPV level continues to decrease and reaches the Low-Low ECCS Actuation Setpoint then escalation to CA2 would be appropriate. For PWRs, i~~If ~~RPV~~Reactor Vessel level continues to decrease and reaches the Bottom ID of the ~~RCS~~PCS Loop, (616 ft. 5.5 in. elevation [Ref. 5, 8]), then escalation to CA2 would be appropriate. Note that the Bottom ID of the ~~RCS~~PCS Loop Setpoint ~~should be the level equal~~corresponds to the bottom of the ~~RPV~~Reactor Vessel loop penetration (not the low point of the loop).

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

PNP Basis Reference(s):

1. PPC Containment Sump Level Trends
2. P&ID Primary Coolant System M-201, Sheet 1 (A-8, G-8, F-1, D-1)
3. P&ID Radioactive Waste Treatment System Clean M-210, Sheet 2 (G-7, D-7)
4. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (C-13)" (Window #68)
5. SOP-1B, "Primary Coolant System – Cooldown"
6. VEN-M1-BM, Sheet 28 - RLI Display Panel
7. M-398, Sheet 1005 - Level Setting Diagram RVLMS
8. FSAR Figure 4-2 (Hot Leg ID)

SYSTEM MALFUNCTION

CU3

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Loss of All Offsite Power to Essential Busses for ~~Greater Than~~GREATER THAN 15 Minutes.

Operating Mode Applicability: Cold Shutdown
Refueling

~~Example~~ Emergency Action Level:

CU3.1. ~~a.~~ Loss of all offsite power to ~~(site-specific) transformers~~Vital 2400 VAC busses 1C and 1D for GREATER THAN~~greater than~~ 15 minutes.

AND

~~b.~~ At least ~~(site-specific)[number of 1~~one emergency diesel generator is ~~s are~~ supplying power to ~~each~~either Vital 2400 VAC~~emergency~~ busses 1C or 1D.

Basis:

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

The 2400 VAC system consists of Safeguard Transformer 1-1, Station Power Transformer 1-2, Start Up Transformer 1-2, four 2400 VAC busses (1C, 1D, 1E and Safeguards). Busses 1C and 1D are the VITAL (essential) busses that supply power to engineered safeguards loads. [Ref. 1] Diesel Generator 1-1 will start when an undervoltage is sensed on 2400 Volt Bus 1C and Diesel Generator 1-2 will start when an undervoltage is sensed on 2400 Volt Bus 1D. [Ref. 2]

~~Plants that have the capability to cross-tie AC power from a companion unit may take credit for the redundant power source in the associated EAL for this IC. Inability to effect the cross-tie within 15 minutes warrants declaring a NOUE.~~

PNP Basis Reference(s):

1. FSAR Section 8.3.2, "Electrical Systems - 2,400 Volt System"
2. FSAR Section 8.4.1, "Electrical Systems - Emergency Generators"

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

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

Operating Mode Applicability: Cold Shutdown
Refueling

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

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

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

Basis:

This IC is included as an ~~NOUE~~ UE because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or ~~RCS~~PCS inventory. Since the ~~RCS~~PCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered. In cold shutdown the decay heat available to raise ~~RCS~~PCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power. Entry into the refueling mode procedurally may not occur for typically ~~100~~ 87 hours ~~(site-specific)~~ or longer after the reactor has been shutdown [Ref 1]. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the ~~RPV~~Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). In addition, the operators should be able to monitor ~~RCS~~PCS temperature and ~~RPV~~Reactor Vessel level so that escalation to the alert level via CA4 or CA1 will occur if required.

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

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

be interrupted. However, if all level and temperature indication were to be lost in either the cold shutdown ~~of~~ or refueling modes, ~~EAL 2~~CU4.2 would result in declaration of a ~~NOUE~~ UE if either temperature or level indication cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert would be via CA2 based on an inventory loss or CA4 based on exceeding its temperature criteria (200 degrees F) [Ref. 2].

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

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

PNP Basis Reference(s):

1. GOP-14, "Shutdown Cooling Operations Attachment 7"
2. Technical Specifications Table 1.1-1

SYSTEM MALFUNCTION

CU5

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Fuel Clad Degradation.

Operating Mode Applicability: Cold Shutdown
Refueling

~~Example~~ Emergency Action Levels: (CU5.1 or CU5.2)

CU5.1. ~~(Site-specific)~~ Any of the following radiation monitors ~~readings~~ with a VALID PPC urgent alarm indicating fuel clad degradation ~~greater than~~ GREATER THAN Technical Specification allowable limits.

Fuel Handling Area Monitor No. 1 - RIA-2316
Fuel Handling Area Monitor No. 2 - RIA-2317
Containment Isolation High Radiation Monitor - RIA-1805
Containment Isolation High Radiation Monitor - RIA-1806
Containment Isolation High Radiation Monitor - RIA-1807
Containment Isolation High Radiation Monitor - RIA-1808

CU5.2. ~~(Site-specific) coolant~~ Coolant ~~sample activity~~sample activity GREATER THAN 1.0 µCi/gm ~~value~~ indicating fuel clad degradation ~~greater than Technical Specification allowable limits~~.

Basis:

This IC is included as a ~~NO~~UE because it is considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. ~~EAL #1~~CU5.1 addresses site-specific radiation monitor readings that provide indication of fuel clad integrity. The Containment Isolation High Radiation Monitors will initiate containment isolation on 2 out of 4 coincidence logic at a reading of 10 R/hr [Ref. 1]. Assured effectiveness of these monitors has been verified by actual testing for the case of core damage assuming approximately 1% failed fuel without LOCA [Ref. 2]. A special test conducted in 1980 in response to NUREG-0737 verified that with approximately 1% failed fuel the Containment Isolation High Radiation monitors would indicate 4.5 Rem/hour [Ref. 3]. The PPC urgent alarm is set at one tenth of that value, 0.45 Rem/hour, indicating fuel clad degradation [Ref. 4]. The Fuel Handling Area Monitors will initiate containment isolation on 1 out of 1 coincident logic at a reading above the expected background for planned operations including movement of the reactor vessel head or internals [Ref. 5]. A calculation performed in 2004 verified that with approximately 1.0 µCi/gm activity dispersed in the flooded Reactor Cavity during refueling operations the Fuel Handling Area Radiation monitors would indicate 4.8 Rem/hour [Ref. 6]. The PPC urgent alarm is set at one tenth of that value, 0.48 Rem/hour, indicating fuel clad degradation [Ref. 7].

~~EAL #2~~ CU5.2 addresses coolant samples exceeding coolant technical specifications for iodine spike. The Technical Specification safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the 10 CFR 100 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit of 1.0 µCi/gm is permissible, if the activity levels do not exceed the limit of 40 µCi/gm for more than 48 hours.

This is acceptable because of the low probability of an SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

Although the Technical Specification is applicable for modes 1, 2 and 3 (when PCS Temperature GREATER THAN OR EQUAL TO 500 degrees F), it is appropriate that this EAL be applicable in cold shutdown and refueling modes, as it indicates a potential degradation in the level of safety of the plant [Ref. 8, 9].

PNP Basis Reference(s):

1. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (EC-13)" (Window 63)
2. FSAR Chapter 7, "Instrumentation and Controls", Section 7.3, Engineered Safeguards Controls, Step 7.3.3.3, "Design Analysis"
3. Consumers Power Company, Response to NUREG-0737, December 19, 1980 (Item II.E.4.3 - Special Test of April 15, 1980)
4. PPC Containment Isolation High Radiation Monitor Urgent Alarm Setpoints
5. Logic Diagram E-17, Sheet 7, "Containment High Radiation" (G-2, G-7)
6. EA-JLV-04-02R0, "Approximate Containment Radiation Monitor Response to Refueling Cavity Water Activity"
7. PPC Fuel Handling Area Radiation Monitor Urgent Alarm Setpoints
8. Technical Specifications 3.4.16, "PCS Specific Activity"
9. Technical Specifications B 3.4.16, "PCS Specific Activity" (Applicable Safety Analysis)

SYSTEM MALFUNCTION

CU6

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.

Operating Mode Applicability: Cold Shutdown
Refueling

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

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

Table C-1 Onsite Communications Systems
• Telephone system • Onsite/offsite radio system • Public address system

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

Table C-2 Offsite Communications Systems
• Telephone system • Power failure phones • FTS phone system • Satellite phone

Basis:

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

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

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

~~Site-specific list for~~Table C-2 offsite communications loss [Ref. 1, 2, 3]~~ must~~ encompasses the loss of all means of communications with offsite authorities. This ~~should~~ includes the ENS, commercial telephone lines, telecopy transmissions, and dedicated phone systems.

PNP Basis Reference(s):

1. FSAR Section 7.7.8, "Instrumentation and Controls - In-Plant Communication System"
2. SOP-31, "Plant Lighting and Communications"
3. EI-15.2, "Communications Tests"

SYSTEM MALFUNCTION

CU7

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

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

Operating Mode Applicability: Cold Shutdown
Refueling

~~Example~~ Emergency Action Level:

~~1. a.~~ CU7.1 -UNPLANNED Loss of v~~V~~ital DC power to required DC busses based on ~~(site-specific)~~ bus voltage indications LESS THAN 105 VDC.

AND

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

Basis:

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

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

~~(Site-specific)~~LESS THAN 105 VDC bus voltage ~~should be~~is based on the minimum bus voltage necessary for the operation of safety related equipment. ~~This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is usually near the minimum voltage selected when battery sizing is performed. Typically the value for the entire battery set is approximately 105 VDC. For a 60 cell string of batteries the cell voltage 1.75 Volts per cell. For a 58 string battery set the minimum voltage is typically 1.81 Volts per cell.~~

Station Battery #1 and Station Battery #2 have ample capacity to supply required DC loads and preferred AC loads during a complete loss of AC power for at least four hours, assuming neither emergency diesel generator is available. The batteries are designed to furnish their maximum load down to an operating temperature of 70°F without dropping below 105 VDC, and the equipment supplied by the batteries is capable of operating satisfactorily at this voltage rating. 105 VDC represents ~80% of the manufacturers' rating for battery capacity of a nominal 131 VDC. [Ref. 1]

PNP Basis Reference(s):

1. FSAR Section 8.4.2, "Electrical Systems - Station Batteries"

SYSTEM MALFUNCTION

CU8

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

Inadvertent Criticality.

Operating Mode Applicability: Cold Shutdown
Refueling

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

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

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

Basis:

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

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

Escalation would be by Emergency Director Judgment.

PNP Basis Reference(s):

None

SYSTEM MALFUNCTION

CA1

Initiating Condition -- ALERT

Loss of ~~RCS~~PCS Inventory.

Operating Mode Applicability: Cold Shutdown

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

CA1.1. Loss of ~~RCS~~PCS inventory as indicated by ~~RPV~~Reactor Vessel level ~~less than~~LESS THAN 617 ft. 0 in. elevation.~~{site-specific level}.~~
~~(low-low ECCS actuation setpoint) (BWR)~~
~~(bottom ID of the RCS loop) (PWR)~~

CA1.2. ~~a.~~ Loss of ~~RCS~~PCS inventory as indicated by unexplained ~~{site-specific} sump~~ Containment Sump OR Primary System Drain Tank -level ~~increase~~rise

AND

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

Basis:

These ~~example~~ EALs serve as precursors to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further ~~RPV~~Reactor Vessel level decrease and potential core uncovery. The 616 ft. 5.5 in. elevation threshold corresponds to the bottom inside diameter of the ~~RCS~~PCS loop [Ref. 1, 2]. This condition will result in a minimum classification of Alert. ~~The BWR Low-Low ECCS Actuation Setpoint was chosen because it is a standard setpoint at which all available injection systems automatically start.~~ The ~~PWR~~ Bottom ID of the ~~RCS~~PCS Loop Setpoint was chosen because at this level remote ~~RCS~~PCS level indication may be lost and loss of suction to decay heat removal systems has occurred. The Bottom ID of the ~~RCS~~PCS Loop Setpoint ~~should be~~is the level equal to the bottom of the ~~RPV~~Reactor Vessel loop penetration (not the low point of the loop). The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the ~~RCS~~PCS barrier.

~~B~~When Reactor Vessel water level decreases to 616 ft. 5.5 in. elevation, the bottom of the PCS hot leg is uncovered. Reactor Vessel Level Transmitter LT-0105 and Reactor Hot Leg Level Transmitter LT-0106 provide the closest readily available indication of this level. Both transmitters will indicate approximately 0% at a level of 617 ft. 0 in. [Ref 2]

In cold shutdown the decay heat available to raise ~~RCS~~PCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically ~~100~~ 87 hours ~~{site-specific}~~ or longer after the reactor has been shutdown [Ref. 3]. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the ~~RPV~~Reactor Vessel (note that the heatup threat could be lower for

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

In the cold shutdown mode, normal ~~RCS~~PCS level and ~~RPV~~Reactor Vessel level instrumentation systems will normally be available. However, if all level indication were to be lost during a loss of ~~RCS~~PCS inventory event, the operators would need to determine that ~~RPV~~Reactor Vessel inventory loss was occurring by observing sump and tank level changes. Rising containment sump level can be monitored on the PPC [Ref. 4]. Rising Primary System Drain Tank level can be monitored at EC-40, "Radwaste Panel" annunciation for which will alarm both locally and in the Control Room [Ref. 5, 6, 7]. Sump and tank level ~~increase~~rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of ~~RCS~~PCS leakage ~~[Ref. 3]~~. The 15-minute duration for the loss of level indication was chosen because it is half of the CS1 Site Area Emergency EAL duration. The 15-minute duration allows CA1 to be an effective precursor to CS1. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CS1 basis. Therefore this EAL meets the definition for an Alert emergency.

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

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

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

PNP Basis Reference(s):

1. FSAR Figure 4-2 (Hot Leg ID)
2. SOP-1B, "Primary Coolant System – Cooldown"
3. GOP-14, "Shutdown Cooling Operations" Attachment 7
4. PPC Containment Sump Level Trends
5. P&ID Primary Coolant System M-201, Sheet 1 (A-8, G-8, F-1, D-1)
6. P&ID Radioactive Waste Treatment System Clean M-210, Sheet 2 (G-7, D-7)
7. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (C-13)" (Window #68)

SYSTEM MALFUNCTION

CA2

Initiating Condition -- ALERT

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

Operating Mode Applicability: Refueling

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

CA2.1. Loss of ~~RPV~~Reactor Vessel inventory as indicated by ~~RPV~~Reactor Vessel level ~~less than~~LESS THAN 617 ft. 0 in. elevation.~~{site-specific level}~~

CA2.2. ~~a.~~ Loss of ~~RPV~~Reactor Vessel inventory as indicated by unexplained Containment Sump OR Primary System Drain Tank ~~A{site-specific} sump and tank~~ level ~~increase~~rise

AND

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

Basis:

These ~~example~~ EALs serve as precursors to a loss of heat removal. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further ~~RPV~~Reactor Vessel level decrease and potential core uncovery. 616 ft. 5.5 in. el. threshold corresponds to the bottom inside diameter of the ~~RCS~~PCS loop [Ref. 1, 2]. This condition will result in a minimum classification of Alert. ~~The BWR Low-Low ECCS Actuation Setpoint was chosen because it is a standard setpoint at which all available injection systems automatically start.~~ The Bottom ID of the ~~RCS~~PCS Loop Setpoint was chosen because at this level remote ~~RCS~~PCS level indication may be lost and loss of suction to decay heat removal systems may occur. The Bottom ID of the ~~RCS~~PCS Loop Setpoint ~~should be~~is the level equal to the bottom of the ~~RPV~~Reactor Vessel loop penetration (not the low point of the loop). The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the ~~RCS~~PCS barrier.

~~□~~When Reactor Vessel water level decreases to 616 ft. 5.5 in. elevation, the bottom of the PCS hot leg is uncovered. Reactor Vessel Level Transmitter LT-0105 and Reactor Hot Leg Level Transmitter LT-0106 provide the closest reliable indication of this level. Both transmitters will indicate approximately 0% at a level of 617 ft. 0 in. [Ref 2]

In cold shutdown the decay heat available to raise ~~RCS~~PCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically ~~100~~ 87 hours ~~{site-specific}~~ or longer after the reactor has been shutdown [Ref. 3]. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the ~~RPV~~Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above

forms the basis for needing both a cold shutdown specific IC (CA1) and a refueling specific IC (CA2).

In the refueling mode, normal means of ~~RPV~~Reactor Vessel level indication may not be available. Redundant means of ~~RPV~~Reactor Vessel level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of ~~RCS~~PCS inventory event, the operators would need to determine that ~~RPV~~Reactor Vessel inventory loss was occurring by observing sump and tank level changes. Rising containment sump level can be monitored on the PPC [Ref. 4]. Rising Primary System Drain Tank level can be monitored at EC-40, "Radwaste Panel" annunciation for which will alarm both locally and in the Control Room [Ref. 5, 6, 7]. Sump and tank level ~~increase~~rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of ~~RCS~~PCS leakage. The 15-minute duration for the loss of level indication was chosen because it is half of the CS2 Site Area Emergency EAL duration. The 15-minute duration allows CA2 to be an effective precursor to CS2. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CS2 basis. Therefore this EAL meets the definition for an Alert.

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

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

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

PNP Basis Reference(s):

1. FSAR Figure 4-2 (Hot Leg ID)
2. SOP-1B, "Primary Coolant System – Cooldown"
3. GOP-14, "Shutdown Cooling Operations Attachment 7"
4. PPC Containment Sump Level Trends
5. P&ID Primary Coolant System M-201, Sheet 1 (A-8, G-8, F-1, D-1)
6. P&ID Radioactive Waste Treatment System Clean M-210, Sheet 2 (G-7, D-7)
7. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (C-13)" (Window #68)

SYSTEM MALFUNCTION

CA3

Initiating Condition -- ALERT

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

Operating Mode Applicability: Cold Shutdown
Refueling
Defueled

~~Example~~ Emergency Action Level:

CA3.1. ~~a.~~ Loss of all offsite power to ~~(site-specific) transformers~~ both Vital 2400 VAC busses 1C and 1D.

AND

~~b.~~ Failure of ~~(site-specific)all~~both emergency diesel generators to supply power to Vital 2400 VAC ~~emergency~~ busses.

AND

~~c.~~ Failure to restore power to at least one Vital 2400 VAC ~~emergency~~ bus within 15 minutes from the time of loss of both offsite and onsite AC power.

Basis:

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

The 2400 VAC system consists of Safeguard Transformer 1-1, Station Power Transformer 1-2, Start Up Transformer 1-2, four 2400 VAC busses (1C, 1D, 1E and Safeguards). Busses 1C and 1D are the VITAL (essential) busses that supply power to engineered safeguards loads. [Ref. 1] Diesel Generator 1-1 will start when an undervoltage is sensed on 2400 Volt Bus 1C and Diesel Generator 1-2 will start when an undervoltage is sensed on 2400 Volt Bus 1D. [Ref. 2]

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of AC power to essential busses. Even though an essential bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus then the bus should not be considered operable.

PNP Basis Reference(s):

1. FSAR Section 8.3.2, "Electrical Systems - 2,400 Volt System"
2. FSAR Section 8.4.1, "Electrical Systems - Emergency Generators"

SYSTEM MALFUNCTION

CA4

Initiating Condition -- ALERT

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

Operating Mode Applicability: Cold Shutdown
Refueling

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

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

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

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

Basis:

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

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

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

Containment barrier to fission product release is established. Note 1 indicates that ~~EAL 2~~CA4.2 is not applicable if actions are successful in restoring an ~~RCS~~PCS heat removal system to operation and ~~RCS~~PCS temperature is being reduced within the 20 minute time frame.

~~EAL 3~~CA4.3 addresses complete loss of functions required for core cooling for >GREATER THAN 60 minutes during refueling and cold shutdown modes when ~~RCS~~PCS integrity is established. As in ~~EAL 1~~CA4.1 and ~~2~~CA4.2, ~~RCS~~PCS integrity should be considered to be in place when the ~~RCS~~PCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). The status of CONTAINMENT CLOSURE in this EAL is immaterial given that the ~~RCS~~PCS is providing a high pressure barrier to fission product release to the environment. The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety. The ~~{site-specific}~~10 psig pressure ~~increase~~rise covers situations where, due to high decay heat loads, the time provided to restore temperature control, should be less than 60 minutes. Digital PZR Pressure Indicator PI-0104 narrow range (0 to 600 psia) is capable of measuring pressure to less than 25 psia. [Ref. 2]~~. The RCS pressure setpoint chosen should be 10 psig or the lowest pressure that the site can read on installed Control Board instrumentation that is equal to or greater than 10 psig.~~ Note 1 indicates that ~~EAL 3~~CA4.3 is not applicable if actions are successful in restoring an ~~RCS~~PCS heat removal system to operation and ~~RCS~~PCS temperature is being reduced within the 60 minute time frame assuming that the ~~RCS~~PCS pressure ~~increase~~rise has remained less than the site specific pressure value.

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

~~For PWRs, t~~This IC and its associated EALs are based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." A number of phenomena such as pressurization, vortexing, steam generator U-tube draining, ~~RCS~~PCS level differences when operating at a mid-loop condition, decay heat removal system design, and level instrumentation problems can lead to conditions where decay heat removal is lost and core uncovery can occur. NRC analyses show that sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary UNPLANNED excursion above 200 degrees F [Ref. 3] when the heat removal function is available.

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

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

PNP Basis Reference(s):

1. GOP-14, "Shutdown Cooling Operations", Attachment 1 - Terms and Definitions
2. SOP-1B, "Primary Coolant System – Cooldown"
3. Technical Specifications Table 1.1-1

SYSTEM MALFUNCTION

CS1

Initiating Condition -- SITE AREA EMERGENCY

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

Operating Mode Applicability: Cold Shutdown

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

CS1.1. With CONTAINMENT CLOSURE not established:

a. ~~RPV~~ Reactor Vessel inventory as indicated by ~~RPV~~ Reactor Vessel level LESS THAN ~~{site-specific level}~~616 ft. 6 in. elevation.

OR

b. ~~RPV~~Reactor Vessel level cannot be monitored for ~~>~~ GREATER THAN 30 minutes with a loss of ~~RPV~~Reactor Vessel inventory as indicated by unexplained Containment Sump OR Primary System Drain Tank ~~A and or ank~~ level ~~increase~~rise.

CS1.2. With CONTAINMENT CLOSURE established:

a. ~~RPV~~Reactor Vessel inventory as indicated by ~~RPV~~Reactor Vessel level ~~less than~~LESS THAN 614 ft. 0 in. elevation.~~TOAF~~

OR

b. ~~RPV~~Reactor Vessel level cannot be monitored for ~~>~~ GREATER THAN 30 minutes with a loss of ~~RPV~~Reactor Vessel inventory as indicated by either:
   - Unexplained ~~{site-specific}~~ ~~s~~Containment Sump OR Primary System Drain Tank ~~and tank~~ level ~~increase~~rise
   - Erratic Source Range Monitor Indication

Basis:

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

In cold shutdown the decay heat available to raise ~~RCS~~PCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically ~~100~~ 87 hours ~~{site-specific}~~ or longer after the reactor has been shutdown. [Ref. 1] Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the ~~RPV~~Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above

forms the basis for needing both a cold shutdown specific IC (CS1) and a refueling specific IC (CS2). ~~[Ref. 3, 4]~~

In the cold shutdown mode, normal ~~RCS~~PCS level and reactor vessel level ~~indication~~ monitoring systems (RVL~~I~~MS) will normally be available. However, if all level indication were to be lost during a loss of ~~RCS~~PCS inventory event, the operators would need to determine that ~~RPV~~Reactor Vessel inventory loss was occurring by observing sump and tank level changes. Rising containment sump level can be monitored on the PPC [Ref. 2]. Rising Primary System Drain Tank level can be monitored at EC-40, "Radwaste Panel" annunciation for which will alarm both locally and in the Control Room [Ref. 3, 4, 5]. Sump and tank level ~~increase~~rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of ~~RCS~~PCS leakage. Additionally, post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors can be used as a tool for making such determinations. Two of the source range neutron flux indicators (NI-1/3A and NI-2/4A) provide control room indication [Ref. 6]. Another source range neutron flux indicator is located at Auxiliary Hot Shutdown Monitoring Panel C-150A (NI-1/3C) [Ref. 7]. Visual and audible indication is available in the Control Room and audible indication in Containment when in service. [Ref. 8].

These ~~example~~ EALs are based on concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*, SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*, NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*, and, NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.* A number of variables, ~~(BWRs - e.g., such as initial vessel level, or shutdown heat removal system design)~~ (PWRs - e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, ~~RCS~~PCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining) can have a significant impact on heat removal capability challenging the fuel clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30-minutes was chosen.

~~If a PWRs RVLIS~~ For CS1.1 RVLMS is unable to distinguish 6" below the bottom ID of the ~~RCS~~ PCS loop penetration. The closest indication of this level is provided by RVLMS UGS Region Sensor #7 red light on, ~40 in. above the fuel alignment plate at 616 ft. 6 in. el. [Ref. 9, 10, 11]. This indication allows clear escalation from CA1.1 and maintains the 6 in. difference in PCS elevation between the Alert and Site Area Emergency classification prescribed by NEI 99-01. ~~, then the first observable point below the bottom ID of the loop should be chosen as the setpoint.~~ If ~~a RVLIS~~RVLMS is not available such that the ~~PWR~~ EAL setpoint cannot be determined, then ~~EAL~~ CS1.1.b should be used to determine if the IC has been met.

For CS1.2 when Reactor Vessel water level drops to 613 ft. 2 in. elevation (Top Of Active Fuel), core uncovery is about to occur. The closest indication of this level is provided by RVLMS UGS Region Sensor #8 red light on, ~11 in. above the fuel alignment plate at 614 ft. 0 in. el. [Ref. 9, 10, 11].

The 30-minute duration allowed when CONTAINMENT CLOSURE [Ref. 12] is established allows sufficient time for actions to be performed to recover needed cooling equipment and is considered to be conservative given that level is being monitored via CS1 and CS2. ~~For PWRs the~~ Eeffluent release is not expected with closure established. ~~For BWRs releases would be monitored and escalation would be via Category A ICs if required.~~

Thus, ~~for both PWR and BWR~~ declaration of a Site Area Emergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via CG1 (Loss of

~~RPV~~Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the ~~RPV~~Reactor Vessel) or radiological effluent IC ~~AG1~~ RG1 (Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology).

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

PNP Basis Reference(s):

1. GOP-14, "Shutdown Cooling Operations", Attachment 7
2. PPC Containment Sump Level Trends
3. P&ID Primary Coolant System M-201, Sheet 1 (A-8, G-8, F-1, D-1)
4. P&ID Radioactive Waste Treatment System Clean M-210, Sheet 2 (G-7, D-7)
5. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (C-13)" (Window #68)
6. SOP-35, "Neutron Monitoring System"
7. EA-APR-95-001, "Appendix R Safe Shutdown Equipment List and Logic Diagrams"
8. GOP-11, "Refueling Operation and Fuel Handling", Attachment 2 - Refuel Handling Operation Shift Checklist
9. VEN-M1-BM, Sheet 28, "RLI Display Panel"
10. M-398, Sheet 1005, "Level Setting Diagram RVLMS"
11. EOP Setpoint Basis (Top Of Active Fuel region)
12. GOP-14, "Shutdown Cooling Operations", Attachment 1 - Terms and Definitions

SYSTEM MALFUNCTION

CS2

Initiating Condition -- SITE AREA EMERGENCY

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

Operating Mode Applicability: Refueling

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

CS2.1. With CONTAINMENT CLOSURE not established:

a. ~~RPV~~ Reactor Vessel inventory as indicated by ~~RPV~~ Rector Vessel level LESS THAN ~~{site-specific level}~~616 ft. 6 in. elevation.

OR

b. ~~RPV~~Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by ~~one or more~~any of the following:
- Containment High Range Radiation Monitor reading ~~>~~GREATER THAN ~~{site-specific} setpoint~~40 R/hr
- Erratic Source Range Monitor Indication

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

CS2.2. With CONTAINMENT CLOSURE established

a. ~~RPV~~Reactor Vessel inventory as indicated by ~~RPV~~Reactor Vessel level ~~less than~~LESS THAN 614 ft. 0 in. elevation.~~TOAF~~

OR

b. ~~RPV~~Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by ~~one or more~~any of the following:
- Containment High Range Radiation Monitor reading ~~>~~GREATER THAN ~~{site-specific} setpoint~~40 R/hr
- Erratic Source Range Monitor Indication

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

Basis:

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

In cold shutdown the decay heat available to raise ~~RCS~~PCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically ~~100~~ 87 hours ~~{site-specific}~~ or longer after the reactor has been shutdown. [Ref. 1] Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the ~~RPV~~Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CS1) and a refueling specific IC (CS2).

These ~~example~~ EALs are based on concerns raised by Generic Letter 88-17, *Loss of Decay Heat* Removal, SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*, NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States* , and, NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.* A number of variables, ~~(BWRs – e.g., such as initial vessel level, or shutdown heat removal system design) (PWRs –~~ (e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, ~~RCS~~PCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining) can have a significant impact on heat removal capability challenging the fuel clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery ~~therefore, conservatively, 30 minutes was chosen~~.

~~If a PWRs RVLIS~~ For CS2.1 RVLMS is unable to distinguish 6" below the bottom ID of the ~~RCS~~ PCS loop penetration. The closest indication of this level is provided by RVLMS UGS Region Sensor #7 red light on, ~40 in. above the fuel alignment plate at 616 ft. 6 in. el. [Ref. 3, 4]. This indication allows clear escalation from CA2.1 and maintains the 6 in. difference in PCS elevation between the Alert and Site Area Emergency classification prescribed by NEI 99-01.~~, then the first observable point below the bottom ID of the loop should be chosen as the setpoint.~~ If ~~a RVLIS~~ RVLMS is not available such that the ~~PWR~~ EAL setpoint cannot be determined, then ~~EAL~~ CS2.1.b should be used to determine if the IC has been met.

For CS2.2 when Reactor Vessel water level drops to 613 ft. 2 in. elevation (Top Of Active Fuel) [Ref. 2], core uncovery is about to occur. The closest indication of this level is provided by RVLMS UGS Region Sensor #8 red light on, ~11 in. above the fuel alignment plate at 614 ft. 0 in. elevation. [Ref. 3, 4]

In Refueling mode, normal PCS level indication (e.g., RVLMS) 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 Containment High Range Radiation monitors RIA-2321 or RIA-2322, or erratic Source Range Monitor indication. The value of 40 Rem/hr on Containment High Range Radiation monitors RIA-2321 or RIA-2322 corresponds to the alert alarm. [Ref. 5, 6] Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and Source Range Monitors can be used as a tool for making such determinations. Two of the source range neutron flux indicators (NI-1/3A and NI-2/4A) provide control room indication [Ref. 7]. Another source range neutron flux indicator is located at Auxiliary Hot Shutdown Monitoring Panel C-150A (NI-1/3C) [Ref. 8]. Visual and audible indication is available in the Control Room and audible indication in Containment when in service. [Ref. 9]

As water level in the ~~RPV~~ Reactor Vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in up-scaled Containment High Range Monitor

indication and possible alarm. [Ref. 5] ~~EAL 1.b and EAL 2.b calculations should be performed to conservatively estimate a site-specific dose rate setpoint indicative of core uncovery (ie., level at TOAF). Additionally, post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.~~

For ~~EAL 2~~CS2.2 in the refueling mode, normal means of ~~RPV~~Reactor Vessel level indication may not be available. Redundant means of ~~RPV~~Reactor Vessel level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

~~For PWRs the e~~Effluent release is not expected with closure established [Ref. 10].
~~For BWRs releases would be monitored and escalation would be via Category A ICs if required.~~

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

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

PNP Basis Reference(s):

1. GOP-14, "Shutdown Cooling Operations", Attachment 7
2. EOP Setpoint Basis (Top Of Active Fuel region)
3. VEN-M1-BM, Sheet 28, "RLI Display Panel"
4. M-398, sheet 1005, "Level Setting Diagram RVLMS"
5. NMC calculation EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4"
6. ARP 33, "Auxiliary Systems Scheme EK-02 (C-11A) Annunciator Nos. 1 and 2" (Windows #13 & #14)
7. SOP-35, "Neutron Monitoring System"
8. EA-APR-95-001, "Appendix R Safe Shutdown Equipment List and Logic Diagrams"
9. GOP-11, "Refueling Operation and Fuel Handling", Attachment 2 - Refuel Handling Operation Shift Checklist
10. GOP-14, "Shutdown Cooling Operations", Attachment 1 - Terms and Definitions

SYSTEM MALFUNCTION

CG1

Initiating Condition -- GENERAL EMERGENCY

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

Operating Mode Applicability: Cold Shutdown
Refueling

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

CG1.1. Loss of ~~RPV~~ Reactor Vessel inventory as indicated by unexplained ~~{site-specific}~~ Containment ~~s~~Sump ~~and~~ OR Primary System Drain ~~t~~Tank level ~~increase~~rise.

AND

~~2.~~ ~~RPV~~ Reactor Vessel Level:

a. LESS THAN 614 ft. 0 in. elevation ~~less than TOAF~~ for ~~>~~GREATER THAN 30 minutes

OR

b. cannot be monitored with indication of core uncovery for ~~>~~GREATER THAN 30 minutes as evidenced by ~~one or more~~any of the following:
   - Containment High Range Radiation Monitor reading ~~>~~GREATER THAN ~~{site-specific} setpoint~~40 R/hr
   - Erratic Source Range Monitor Indication

   ~~Other {site-specific} indications~~

AND

~~3.~~ ~~{Site-specific}~~ indication of Containment challenged as indicated by ~~one or more~~any of the following:
   - ~~Explosive mixture inside containment~~ Containment hydrogen concentration GREATER THAN OR EQUAL TO 6%
   - Containment ~~P~~pressure above ~~{site-specific} value~~ 55 psig
   - CONTAINMENT CLOSURE <u>not</u> established

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

Basis:

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

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

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

~~EAL 2~~This EAL represents the inability to restore and maintain ~~RPV~~Reactor Vessel level to above the top of active fuel. Fuel damage is probable if ~~RPV~~Reactor Vessel level cannot be restored, as available decay heat will cause boiling, further reducing the ~~RPV~~Reactor Vessel level.

~~These example~~is EAL~~s~~ is~~are~~ based on concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*, SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*, NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*, and, NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.* A number of variables~~, (BWRs – e.g., such as initial vessel level, or shutdown heat removal system design) (PWRs –~~(e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, ~~RCS~~PCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining) can have a significant impact on heat removal capability challenging the fuel clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30 minutes was chosen.

Containment Sump or Primary System Drain Tank level changes may be indicative of a loss of PCS inventory. Rising containment sump level can be monitored visually on the PPC [Ref. 1]. Rising Primary System Drain Tank level can be monitored at EC-40, "Radwaste Panel" annunciation for which will alarm both locally and in the Control Room [Ref. 2, 3]. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of PCS leakage.

When Reactor Vessel water level drops to 613 ft. 2 in. elevation (Top Of Active Fuel) [Ref. 4], core uncovery is about to occur. The closest indication of this level is provided by RVLMS UGS Region Sensor #8 red light on, ~11 in. above the fuel alignment plate at 614 ft. 0 in. elevation. Fuel damage is probable if core submergence cannot be restored as available decay heat will cause boiling and further lowers the vessel level. [Ref. 5, 6]

If all means of level monitoring are not available, a Reactor Vessel inventory loss resulting in core uncovery may be detected by the following indirect methods:

- Containment High Range Radiation monitors RIA-2321 or RIA-2322 reading > 40 Rem/hr which corresponds to the alert alarm. [Ref. 7, 8]

- Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors can be used as a tool for making such determinations. Two of the source range neutron flux indicators (NI-1/3A and NI-2/4A) provide control room indication [Ref. 9]. Another source range neutron flux indicator is located at Auxiliary Hot Shutdown Monitoring Panel C-150A (NI-1/3C) [Ref. 10]. Visual and audible indication is available in the Control Room and audible indication in Containment when in service. [Ref. 11]

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

Three conditions are associated with a challenge to containment integrity:

- Standard industry reference documents list the lowest potentially explosive concentration for hydrogen in a standard atmosphere as 6.0%. [Ref. 12]

- The containment design pressure of 55 psig is in excess of that expected from the design basis loss of coolant accident. The threshold is indicative of a loss of both PCS and fuel clad boundaries in that it is not possible to reach this condition without severe core degradation. [Ref. 13]

- ~~In the context of EAL 3,~~ CONTAINMENT CLOSURE [Ref. 14] is the action taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. CONTAINMENT CLOSURE should not be confused with refueling containment integrity as defined in technical specifications. Site shutdown contingency plans typically provide for re-establishing CONTAINMENT CLOSURE following a loss of heat removal or ~~RCS~~PCS inventory functions. If the closure is re-established prior to exceeding the temperature or level thresholds of the ~~RCS~~PCS Barrier and Fuel Clad Barrier EALs, escalation to GE would not occur.

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

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gasses in CONTAINMENT. However, CONTAINMENT monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists.
~~Expanded basis for these assumptions is provided in Appendix C.~~

PNP Basis Reference(s):

1. PPC Containment Sump Level Trends
2. P&ID Primary Coolant System M-201, Sheet 1 (A-8, G-8, F-1, D-1)
3. P&ID Radioactive Waste Treatment System Clean M-210, Sheet 2 (G-7, D-7)
4. EOP Setpoint Basis (Top Of Active Fuel region)
5. VEN-M1-BM, Sheet 28, "RLI Display Panel"
6. M-398, sheet 1005, "Level Setting Diagram RVLMS"
7. NMC calculation EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4"

8. ARP 33, "Auxiliary Systems Scheme EK-02 (C-11A) Annunciator Nos. 1 and 2" (Windows #13 & #14)
9. SOP-35, "Neutron Monitoring System"
10. EA-APR-95-007, "Appendix R Safe Shutdown Equipment List and Logic Diagrams"
11. GOP-11, "Refueling Operation and Fuel Handling", Attachment 2 - Refuel Handling Operation Shift Checklist
12. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident"
13. FSAR Section 5.8.1, "Design of Structures, Systems and Components - Design Basis"
14. GOP-14, "Shutdown Cooling Operations", Attachment 1 - Terms and Definitions

Table E-0

Recognition Category E

Events Related to ISFSI Malfunction

INITIATING CONDITION MATRIX

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

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~~E-~~HEU1

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Damage to a loaded cask CONFINEMENT BOUNDARY.

Operating Mode Applicability: Not applicable

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

EU1.1. Natural phenomena events affecting a loaded cask CONFINEMENT BOUNDARY as indicated by VISIBLE DAMAGE:

- earthquake
- tornado
- flood
- seiche
- lightning

~~(site-specific list)~~

EU1.2. Accident conditions affecting a loaded cask CONFINEMENT BOUNDARY as indicated by VISIBLE DAMAGE:

- tipped over or dropped cask
- FIRE
- EXPLOSION

~~(site-specific list)~~

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

Basis:

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

Since the CONFINEMENT BOUNDARY is not directly accessible for visual inspection, the EAL definition of VISIBLE DAMAGE to the CONFINEMENT BOUNDARY is defined as: damage to the transfer cask, ventilated concrete cask, or horizontal storage module that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of the canister inside the transfer cask, ventilated concrete cask or horizontal storage module. Example damage includes: deformation due to heat, impact, or unplanned movement, denting, penetration, rupture, cracking, or spalling of concrete to expose concrete reinforcing bar, or reduction in depth or configuration of radiation shielding materials.

Surface blemishes (e.g., paint fading, paint chipping, concrete cracks or scratches) are not included in VISIBLE DAMAGE.

For the events of concern here, the wind speed, earthquake intensity, height of loaded transfer cask drop as a result of normal handling or transporting etc. in and of themselves, are not the key issue. The key issue is whether the resultant damage to or loss of the loaded fuel cask CONFINEMENT BOUNDARY leads to the degradation of the fuel during transfer or storage, or poses an operational safety problem with respect to its removal from storage.

~~For EAL #1 and EAL #2, the results of the ISFSI Safety Analysis Report (SAR) per NUREG 1536 or SAR referenced in the cask('s) Certificate of Compliance and the related NRC Safety Evaluation Report should be used to develop the site-specific list of natural phenomena events and accident conditions. These EALs would address 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.).~~

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

PNP Basis Reference(s):

1. Safety Evaluation Report For The Pacific Sierra Nuclear Associates Safety Analysis Report For The Ventilated Storage Cask System
2. "Final Safety Analysis Report For The Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel"

EVENTS RELATED TO ISFSI

~~E-~~HEU2

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

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

Operating Mode Applicability: Not applicable

~~Example~~ Emergency Action Levels:

EU2.1. Security Event as determined from ~~(site-specific)~~ the PNP Security Plan and reported by the ~~(site-specific) s~~Security ~~s~~Shift ~~supervision~~Leader.

Basis:

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

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

PNP Basis Reference(s):
1. Security Plan (Safeguards)

Table ~~5~~-F-~~10~~

Recognition Category F

Fission Product Barrier Degradation

INITIATING CONDITION MATRIX

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

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

NOTES

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

   - The Fuel Clad Barrier and the ~~RCS~~PCS Barrier are weighted more heavily than the Containment Barrier ~~(See Sections 3.4 and 3.8)~~. ~~NO~~UE ICs associated with ~~RCS~~PCS and Fuel Clad Barriers are addressed under System Malfunction ICs.

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

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

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

~~TABLE 5-F-2~~

~~BWR Emergency Action Level~~

~~Fission Product Barrier Reference Table~~

~~Thresholds For LOSS or POTENTIAL LOSS of Barriers*~~

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

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

Fuel Clad Barrier Example EALS		RCS Barrier Example EALS		Containment Barrier Example EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. Primary Coolant Activity Level		1. Drywell Pressure		1. Drywell Pressure	
Coolant Activity GREATER THAN (site-specific)Value	Not Applicable	Drywell pPressure GREATER THAN (site-specific) PSIG not caused by a loss of DW Cooling	Not Applicable	Rapid unexplained decrease in drywell pressure following initial increase OR Drywell pressure response not consistent with LOCA conditions indicating containment breach	Drywell Pressure GREATER THAN (Site-specific value) PSIG and increasing OR Explosive mixture existsDrywell or torus H2
OR		OR		OR	
2. Reactor Vessel Water Level		2. Reactor Vessel Water Level		2. Reactor Vessel Water Level	
Level LESS THAN (site-specific value)	Level LESS THAN (site-specific value)	Level LESS THAN (site-specific value)	Not Applicable	Not Applicable	Primary containment flooding required
		OR		OR	
		3. RCS Leak Rate		3. CNMT Isolation Failure or Bypass	
		(Site-specific) Indication of an unisolable Main Steamline Break	RCS leakage GREATER THAN 50 gpm inside the drywell OR Unisolable primary system leakage outside drywell as indicated by: -area temperature or area radiation alarm	Failure of both valves in any one line to close AND downstream pathway to the environment exists OR Intentional per EOPs OR Unisolable primary system leakage outside drywell as indicated by: -area temperature or area radiation alarm	Not applicable
OR		OR		OR	

~~TABLE 5-F-2~~

~~BWR Emergency Action Level~~

~~Fission Product Barrier Reference Table~~

~~Thresholds For LOSS or POTENTIAL LOSS of Barriers*~~

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

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

Fuel Clad Barrier Example EALS		RCS Barrier Example EALS		Containment Barrier Example EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
3. Drywell Radiation Monitoring		4. Drywell Radiation Monitoring		4. Significant Radioactive Inventory in Containment	
Drywell Radiation monitor reading GREATER THAN(site specific value) R/hr	Not Applicable	Drywell Radiation monitor reading GREATER THAN (site-specific) R/hr	Not Applicable	Not applicable	Drywell Radiation monitor reading GREATER THAN (site specific) R/hr
OR		**OR**		**OR**	
4. Other (Site Specific) Indications		5. Other (Site-Specific) Indications		5. Other (site-specific) Indications	
(Site specific) as applicable	(Site specific) as applicable	(Site-specific) as applicable	(Site-specific) as applicable	(Site specific) as applicable	(Site specific) as applicable
OR		**OR**		**OR**	
5. Emergency Director Judgment		6. Emergency Director Judgment		6. Emergency Director Judgment	
Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier		Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the RCS Barrier		Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment barrier	

~~Basis Information For Table 5-F-2~~
~~BWR Emergency Action Level~~
~~Fission Product Barrier Reference Table~~

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

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

~~1. Primary Coolant Activity Level~~

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

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

~~2. Reactor Vessel Water Level~~

~~The "Loss" EAL (site-specific) value corresponds to the level which is used in EOPs to indicate challenge of core cooling. Depending on the plant this may be top of active fuel or 2/3 coverage of active fuel. This is the minimum value to assure core cooling without further degradation of the clad. The "Potential Loss" EAL is the same as the RCS barrier "Loss" EAL #2 below and corresponds to the (site-specific) water level at the top of the active fuel. Thus, this EAL indicates a "Loss" of RCS barrier and a "Potential Loss" of the Fuel Clad Barrier. This EAL appropriately escalates the emergency class to a Site Area Emergency. If the "Loss" value is also the Top of Active Fuel, the "Potential Loss" value must be a value indicating a higher level also corresponding to a higher level indicated in the RCS barrier "Loss" EAL #2.~~

~~3. Drywell Radiation Monitoring~~

~~The (site-specific) reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the drywell. The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I-131 or the calculated concentration equivalent to the clad damage used in EAL #1 into the drywell 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. This value is higher than that specified for RCS barrier Loss EAL #4. Thus, this EAL indicates a loss of both Fuel Clad barrier and RCS barrier.~~

~~Caution:~~ *~~it is important to recognize that in the event the radiation monitor is sensitive to shine from the reactor vessel or piping, spurious readings will be present and another indicator of fuel clad damage is necessary or compensated for in the threshold value.~~*

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

~~4. Other (Site-Specific) Indications~~

~~This EAL is to cover other (site-specific) indications that may indicate loss or potential loss of the Fuel Clad barrier, including indications from containment air monitors or any other (site-specific) instrumentation.~~

~~5. Emergency Director Judgment~~

~~This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost. (See also IC SG1, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power", for additional information.)~~

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

~~The RCS Barrier is the reactor coolant system pressure boundary and includes the reactor vessel and all reactor coolant system piping up to the isolation valves.~~

~~1. Drywell Pressure~~

~~The (site-specific) drywell pressure is based on the drywell high pressure set point which indicates a LOCA by automatically initiating the ECCS or equivalent makeup system.~~

~~There is no "Potential Loss" EAL corresponding to this item.~~

~~2. Reactor Vessel Water Level~~

~~This "Loss" EAL is the same as "Potential Loss" Fuel Clad Barrier EAL #2. The (site-specific) water level corresponds to the level which is used in EOPs to indicate challenge of core cooling. Depending on the plant this may be top of active fuel or 2/3 coverage of active fuel. This EAL appropriately escalates the emergency class to a Site Area Emergency. Thus, this EAL indicates a loss of the RCS barrier and a Potential Loss of the Fuel Clad Barrier.~~

~~There is no "Potential Loss" EAL corresponding to this item.~~

~~3. RCS Leak Rate~~

~~An unisolable MSL break is a breach of the RCS barrier. Thus, this EAL is included for consistency with the Alert emergency classification. The potential loss of RCS based on leakage is set at a level indicative of a small breach of the RCS but which is well within the makeup capability of normal and emergency high pressure systems. Core uncovery is not a significant concern for a 50 gpm leak, however, break propagation leading to significantly larger loss of inventory is possible. Many BWRs may be unable to measure an RCS leak of this size because the leak would likely increase drywell pressure above the drywell isolation set point. The system normally used to monitor leakage is typically isolated as part of the drywell isolation and is therefore unavailable. If primary system leak rate information is unavailable, other indicators of RCS leakage should be used.~~

~~Potential loss of RCS based on primary system leakage outside the drywell is determined from site-specific temperature or area radiation alarms low setpoint in the areas of the main steam line tunnel, main turbine generator, RCIC, HPCI, etc., which indicate a direct path from the RCS to areas outside primary containment. The indicators should be confirmed to be caused by RCS~~

~~leakage. The area temperature or radiation low alarm setpoints are indicated for this example to enable an Alert classification. An unisolable leak which is indicated by a high alarm setpoint escalates to a Site Area Emergency when combined with Containment Barrier EAL 3 (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.~~

~~**4. Drywell Radiation Monitoring**~~

~~The (site-specific) reading is a value which indicates the release of reactor coolant to the drywell. The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within T/S) into the drywell atmosphere. This reading will be less than that specified for Fuel Clad Barrier EAL #3. Thus, this EAL would be indicative of a RCS leak only. If the radiation monitor reading increased to that value specified by Fuel Clad Barrier EAL #3, fuel damage would also be indicated.~~

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

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

~~**5. Other (Site-Specific) Indications**~~

~~This EAL is to cover other (site-specific) indications that may indicate loss or potential loss of the RCS barrier.~~

~~**6. Emergency Director Judgment**~~

~~This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost. (See also IC SG1, "Prolonged Loss of Offsite Power and Prolonged Loss of All Onsite AC Power", for additional information.)~~

~~**PRIMARY CONTAINMENT BARRIER EXAMPLE EALs:** (1 or 2 or 3 or 4 or 5 or 6)~~

~~The Primary Containment Barrier includes the drywell, the wetwell, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. Containment Barrier EALs are used primarily as discriminators for escalation from an Alert to a Site Area Emergency or a General Emergency.~~

~~**1. Drywell Pressure**~~

~~Rapid unexplained loss of pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure increase indicates a loss of containment integrity. Drywell pressure should increase as a result of mass and energy release into containment from a LOCA. Thus, drywell pressure not increasing under these conditions indicates a loss of containment integrity. This indicator relies on the operators recognition of an unexpected response for the condition and therefore does not have a specific value associated. The unexpected response is important because it is the indicator for a containment bypass condition. The (site-specific) PSIG for potential loss of containment is based on the containment drywell design pressure. Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration~~

~~limit curve exists. This applies to BWRs with Mark III containments, as well as Mark I and II containment designs when they are de-inerted.~~

~~**2. Reactor Vessel Water Level**~~

~~The entry into the Primary Containment Flooding emergency procedure indicates reactor vessel water level can not be restored and that a core melt sequence is in progress. EOPs direct the operators to enter Containment Flooding when Reactor Vessel Level cannot be restored to greater than a Site Specific value (generally 2/3 core height) or is unknown. Entry into Containment Flooding procedures is a logical escalation in response to the inability to maintain reactor vessel level.~~

~~The conditions in this potential loss EAL represent imminent core melt sequences which, if not corrected, could lead to vessel failure and increased potential for containment failure. In conjunction with and an escalation of the level EALs in the Fuel and RCS barrier columns, this EAL will result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third. If the emergency operating procedures have been ineffective in restoring reactor vessel level above the RCS and Fuel Clad Barrier Threshold Values, there is not a "success" path and a core melt sequence is in progress. Entry into Containment flooding procedures is a logical escalation in response to the inability to maintain reactor vessel level.~~

~~Severe accident analysis (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation with the reactor vessel in a significant fraction of the core damage scenarios, and the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow emergency operating procedures to arrest the core melt sequence. Whether or not the procedures will be effective should be apparent within the time provided. The Emergency Director should make the declaration as soon as it is determined that the procedures have been, or will be, ineffective. There is no "loss" EAL associated with this item.~~

~~**3. Containment Isolation Failure or Bypass**~~

~~This EAL is intended to cover the inability to isolate the containment when containment isolation is required. In addition, the presence of area radiation or temperature alarms high setpoint indicating unisolable primary system leakage outside the drywell are covered after a containment isolation. The indicators should be confirmed to be caused by RCS leakage. Also, an intentional venting of primary containment for pressure control per EOPs to the secondary containment and/or the environment is considered a loss of containment. Containment venting for temperature or pressure when not in an accident situation should not be considered.~~

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

~~**4. Significant Radioactive Inventory in Containment**~~

~~The (site-specific) reading is a value which indicates significant fuel damage well in excess of that required for loss of RCS and Fuel Clad. As stated in Section 3.8, a major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant. Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%. Unless there is a~~

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

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

~~5. Other (Site-Specific) Indications~~

~~This EAL is to cover other (site-specific) indications that may indicate loss or potential loss of the containment barrier.~~

~~6. Emergency Director Judgment~~

~~This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost. (See also IC SG1, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power", for additional information.)~~

TABLE ~~5~~-F-41

~~PWR~~ PNP Emergency Action Level

Fission Product Barrier Reference Table

Thresholds For LOSS or POTENTIAL LOSS of Barriers*

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

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY ~~I~~Loss or ANY Potential Loss of Containment	ANY ~~L~~loss or ANY Potential Loss of EITHER Fuel Clad ~~or~~OR ~~RCS~~PCS	Loss or Potential Loss of ANY ~~T~~two Barriers	Loss of ANY ~~T~~two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier ~~Example~~ EALS		RCSPCS Barrier ~~Example~~ EALS		Containment Barrier ~~Example~~ EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. Critical Safety Function Status		**1. Critical Safety Function Status**		**1. Critical Safety Function Status**	
~~Core Cooling Red~~ Not Applicable	~~Core Cooling Orange OR Heat Sink Red~~ Not Applicable	Not Applicable	~~RCS Integrity Red OR Heat Sink Red~~ Not Applicable	Not Applicable	~~Containment Red~~ Not Applicable
OR		**OR**		**OR**	
2. Primary Coolant Activity Level		**2. ~~RCS~~PCS Leak Rate**		**2. Containment Pressure**	
~~Coolant Activity GREATER THAN (site-specific) Value~~Dose rate value for primary coolant GREATER THAN 1 R/hr	Not Applicable	GREATER THAN available makeup capacity ~~as~~ indicated by ~~a loss of~~ ~~RCS~~PCS subcooling LESS THAN 25 degrees F based on average of qualified CETs	Unisolable leak exceeding ~~the capacity of one charging pump in the normal charging mode~~50 gpm	Rapid unexplained ~~decrease~~ lowering following initial ~~increase~~rise OR Containment pressure or sump level response not consistent with LOCA conditions	~~(Site-specific)~~55 ~~PSIG~~ psig and ~~increasing~~ rising OR Containment hydrogen concentration GREATER THAN OR EQUAL TO 6%~~Explosive mixture exists~~ OR Containment p~~P~~ressure ~~greater~~GREATER ~~than containment depressurization actuation setpoint~~ THAN 4 psig with ~~less than~~LESS THAN one full train of depressurization equipment operating
OR				**OR**	
3. Core Exit Thermocouple Readings				**3. Core Exit ~~Themocouple~~Thermocouple Reading**	

TABLE ~~5~~-F-41

~~PWR~~-PNP Emergency Action Level

Fission Product Barrier Reference Table

Thresholds For LOSS or POTENTIAL LOSS of Barriers*

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

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

Fuel Clad Barrier Example ~~EALS~~		RCSPCS Barrier Example ~~EALS~~		Containment Barrier Example ~~EALS~~	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
GREATER THAN 1200 ~~(site-specific)~~ degrees F	GREATER THAN ~~(site-specific)~~ 700 degrees F			Not applicable	Core exit thermocouples in excess of 1200 degrees F and restoration procedures not effective within 15 minutes; ~~or,~~ OR Core exit thermocouples in excess of 700 degrees F with reactor vessel level below 614 ft. 0 in. elevation ~~top of active fuel~~ and restoration procedures not effective within 15 minutes

TABLE ~~5~~-F-41

~~PWR~~-PNP Emergency Action Level

Fission Product Barrier Reference Table

Thresholds For LOSS or POTENTIAL LOSS of Barriers*

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

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY ~~L~~Loss or ANY Potential Loss of Containment	ANY ~~L~~Loss or ANY Potential Loss of EITHER Fuel Clad ~~or~~OR ~~RCS~~PCS	Loss or Potential Loss of ANY ~~T~~two Barriers	Loss of ANY ~~T~~two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier ~~Example~~ EALS		RCSPCS Barrier ~~Example~~ EALS		Containment Barrier ~~Example~~ EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
OR		**OR**		**OR**	
4. Reactor Vessel Water Level		**3. SG Tube Rupture**		**4. SG Secondary Side Release with P-to-S Leakage**	
Not Applicable	Level LESS ~~than~~THAN ~~(site-specific) value~~ (614 ft. 0 in. elevation	SGTR that results in an ECCS (SI) Actuation	Not Applicable	RUPTURED S/G is also FAULTED outside of containment OR Primary-to-Secondary leakrate ~~greater than~~ GREATER THAN 10 gpm with nonisolable steam release from affected S/G to the environment	Not applicable
				OR	
				5. CNMT Isolation Valves Status After CNMT Isolation	
				Containment Isolation Valve(s) not closed AND ~~D~~downstream pathway to the environment exists after containment isolation	Not Applicable
OR		**OR**		**OR**	
5. Containment Radiation Monitoring		**4. Containment Radiation Monitoring**		**6. Significant Radioactive Inventory in Containment**	

TABLE ~~5~~-F-41

~~PWR~~-PNP Emergency Action Level

Fission Product Barrier Reference Table

Thresholds For LOSS or POTENTIAL LOSS of Barriers*

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

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY ILoss or ANY Potential Loss of Containment	ANY LIoss or ANY Potential Loss of EITHER Fuel Clad ~~or~~OR ~~RCS~~PCS	Loss or Potential Loss of ANY Ttwo Barriers	Loss of ANY Ttwo Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier ~~Example~~ EALS		RCSPCS Barrier ~~Example~~ EALS		Containment Barrier ~~Example~~ EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
Containment High Range Radiation Monitor~~Containment rad monitor~~ reading GREATER THAN ~~(site-specific)~~2,000 R/hr as indicated on RIA-2321 and RIA-2322	Not Applicable	Containment High Range Radiation Monitor~~Containment rad monitor~~ reading GREATER THAN ~~(site-specific)~~200 R/hr as indicated on RIA-2321 and RIA-2322	Not Applicable	Not Applicable	Containment High Range Radiation Monitor~~Containment rad monitor~~ reading GREATER THAN ~~(site-specific)~~20,000 R/hr as indicated on RIA-2321 and RIA-2322

TABLE ~~5~~-F-41

~~PWR~~-PNP Emergency Action Level

Fission Product Barrier Reference Table

Thresholds For LOSS or POTENTIAL LOSS of Barriers*

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

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY ~~I~~Loss or ANY Potential Loss of Containment	ANY Lloss or ANY Potential Loss of EITHER Fuel Clad ~~or~~OR ~~RCS~~PCS	Loss or Potential Loss of ANY Ttwo Barriers	Loss of ANY Ttwo Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier ~~Example~~ EALS

LOSS **POTENTIAL LOSS**

OR

6. Other ~~(Site-Specific)~~ Indications

~~(Site-specific)~~ as ~~applicable~~Not Applicable ~~(Site-specific)~~ as ~~applicable~~ Not Applicable

OR

7. Emergency Director Judgment

Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier

~~RCS~~PCS Barrier ~~Example~~ EALS

LOSS **POTENTIAL LOSS**

OR

5. Other ~~(Site-Specific)~~ Indications

~~(Site-specific)~~ as ~~applicable~~Not Applicable ~~(Site-specific)~~ as ~~applicable~~ Not Applicable

OR

6. Emergency Director Judgment

Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the ~~RCS~~PCS Barrier

Containment Barrier ~~Example~~ EALS

LOSS **POTENTIAL LOSS**

OR

7. Other ~~(site-specific)~~ Indications

~~(Site-specific)~~ as ~~applicable~~Not Applicable ~~(Site-specific)~~ as ~~applicable~~Not Applicable

OR

8. Emergency Director Judgment

Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment barrier

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

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

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

1. Critical Safety Function Status

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

~~Core Cooling – ORANGE indicates subcooling has been lost and that some clad damage may occur. Heat Sink – RED indicates the ultimate heat sink function is under extreme challenge and thus these two items indicate potential loss of the Fuel Clad Barrier.~~

~~Core Cooling – RED indicates significant superheating and core uncovery and is considered to indicate loss of the Fuel Clad Barrier.~~Not applicable

2. Primary Coolant Activity Level

PNP is unable to analyze a primary coolant sample of 300 uci/gm due to high dose rates experienced at the NSSS sample panel. Therefore, a dose rate is used in lieu of a sample to definitively indicate fuel clad degradation. The dose rate value used for the primary coolant is specified in accordance with Emergency Implementing Procedure, EI-7.0, "Emergency Post Accident Sampling and Determination of Fuel Failure Using Dose Rates," a sample is taken to measure contact dose rates at the Failed Fuel Survey Point on the PCS sample line. [Ref. 6] The sample is taken to measure contact dose rates at the Failed Fuel Survey Point on the PCS sample line. ~~This (site-specific) value corresponds tois 300 μCi/gm I_{131} equivalent. Assessment by the NUMARC EAL Task Force indicates that this amount of coolant activity~~ The site dose rate value within the first few hours for the primary coolant of approximately 1 R/hr at the NSSS sample sink is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost. [Ref. 6]~~The value expressed can be either in mR/hr observed on the sample or as μCi/gm results from analysis.~~

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

3. Core Exit Thermocouple Readings

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

Core Exit Thermocouples (CETs) provide an indirect indication of fuel cladding temperature by measuring the temperature of the primary coolant that leaves the core region. The "Loss" threshold temperature of 1200 degrees F is consistent with CEOG Generic Accident Management Guidelines, "Phase 1.0 - Initial Diagnosis," for core exit temperature. Although clad rupture due to

high temperature is not expected for CET readings less than the threshold, temperatures of this magnitude signal severe superheating of the primary coolant and core uncovery. [Ref. 3]~~The "Loss" EAL (site-specific) reading should correspond to significant superheating of the coolant. This value typically corresponds to the temperature reading that indicates core cooling – RED in Fuel Clad Barrier EAL #1 which is usually about 1200 degrees F.~~

For the "Potential Loss" threshold, CET reading significantly above the saturation temperature for the existing PCS pressure indicates a superheat condition. 700 degrees F CET temperature is therefore indicative of a potential fuel clad loss. [Ref. 2, 12] Superheat is a valid indication of a potential Fuel Cladding barrier loss condition. ~~The "Potential Loss" EAL (site-specific) reading should correspond to loss of subcooling. This value typically corresponds to the temperature reading that indicates core cooling – ORANGE in Fuel Clad Barrier EAL #1 which is usually about 700 to 900 degrees F.~~

4. Reactor Vessel Water Level

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

For the "Potential Loss" threshold, the RVLMS is based on a heated junction thermocouple system. The thermocouple system measures primary coolant liquid inventory with discrete sensors located at different levels within a separator tube ranging from the fuel alignment plate (i.e., near the top of active fuel) to the top of the Reactor Vessel head. The basic principle of system operation is detection of a temperature difference between heated and unheated thermocouples. [Ref. 18, 19]

Reactor Vessel water level below the top of the core may lead to a Severe Accident Management Guideline "Badly Damaged" condition. The badly damaged descriptor signifies possible core overheating to the point of clad rupture. [Ref. 3] RVLMS is the instrumentation used to indicate reactor vessel level. The lowest indication on the RVLMS is 11 inches above the bottom of the fuel alignment plate, which is approximately the top of the active fuel. RVLMS reading less than 11 inches above the bottom of the fuel alignment plate, elevation 614 ft 0 in., therefore, signals inadequate coolant inventory, loss of subcooling and the occurrence of possible fuel cladding damage. [Ref. 1, 3, 19, 20],
~~The (site-specific) value for the "Potential Loss" EAL corresponds to the top of the active fuel. For sites using CSFSTs, the "Potential Loss" EAL is defined by the Core Cooling – ORANGE path. The (site-specific) value in this EAL should be consistent with the CSFST value.~~

5. Containment Radiation Monitoring

The ~~(site-specific)~~2,000 R/hr reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading ~~should be~~is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I-131 into the containment atmosphere. [Ref. 4] Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage. This value is higher than that specified for ~~RCS~~PCS barrier Loss EAL #4. Thus, this EAL indicates a loss of both the fuel clad barrier and a loss of ~~RCS~~PCS barrier.

Containment radiation is indicated on Containment High Range Radiation Monitors, RIA-2321 and RIA-2322. The Containment High Range Radiation Monitors high alarm at 400 R/hr. [Ref. 5]

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

6. Other ~~(Site-Specific)~~ Indications

There are no "Loss" or potential loss EALs associated with this item.

~~This EAL is to cover other (site-specific) indications that may indicate loss or potential loss of the Fuel Clad barrier, including indications from containment air monitors or any other (site-specific) instrumentation.~~

7. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences. [As discussed in NEI 99-01 rev. 4]

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

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

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to EOP-9.0, "Functional Recovery." The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations. [Ref. 7, 8]

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

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

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

1. Critical Safety Function Status

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

~~There is no "Loss" EAL associated with this item.~~Not applicable

2. ~~RCS~~PCS Leak Rate

The "Loss" EAL addresses conditions where leakage from the ~~RCS~~PCS 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~~PCS pressure and inventory against the mass loss through the leak. Subcooling margin greater than or equal to 25 degree F ensures the fluid surrounding the core is sufficiently cooled and provides margin for reestablishing flow should subcooling deteriorate when SIS flow is secured. [Ref. 2, 11, 15, 21]

The "Potential Loss" EAL is based on the inability to maintain normal liquid inventory within the ~~Reactor Coolant System~~Primary Coolant System (~~RCS~~PCS) by normal operation of the Chemical and Volume Control System (CVCS). which is considered as one ~~centrifugal~~ charging pump discharging to the charging header. ~~A second charging pump being required is indicative of a substantial RCS leak.~~ [Ref. 9] The normal operating charging pump is variable speed, with a capacity of 33 to 53 gpm. The other two charging pumps are fixed speed, with a lower capacity of 40 gpm each. ~~For plants with~~PNP has low capacity charging pumps, therefore a 50 gpm leak rate value ~~may be~~is used to indicate the Potential Loss. [Ref. 2, 9,14]

3. SG Tube Rupture

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

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

There is no "Potential Loss" EAL.

4. Containment Radiation Monitoring

The ~~(site-specific)~~200 R/hr reading is a value which indicates the release of reactor coolant to the containment. The reading ~~should be~~is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within ~~T/S~~Technical Specifications) into the containment atmosphere. [Ref. 4] This reading ~~will be~~is less than that specified for Fuel Clad Barrier EAL #5. Thus, this EAL would be indicative of a ~~RCS~~PCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad Barrier EAL #5, fuel damage would also be indicated.

Containment radiation is indicated on Containment High Range Radiation Monitors, RIA-2321 and RIA-2322. The Containment High Range Radiation Monitors high alarm at 400 R/hr. [Ref. 5] ~~However, if the site-specific physical location of the containment radiation monitor is such that radiation from a cloud of released RCS gases could not be distinguished from radiation from nearby piping and components containing elevated reactor coolant activity, this EAL should be omitted and other site specific indications of RCS leakage substituted.~~

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

5. Other ~~(Site-Specific)~~ Indications

There are no "Loss" or potential loss EALs associated with this item.

~~This EAL is to cover other (site-specific) indications that may indicate loss or potential loss of the RCS barrier, including indications from containment air monitors or any other (site-specific) instrumentation.~~

6. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the ~~RCS~~PCS barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences. [As discussed in NEI 99-01 rev. 4]

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

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

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to EOP-9.0, "Functional Recovery." The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations. [Ref. 7, 8]

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

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

The Containment Barrier includes the containment building, its connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve.

1. Critical Safety Function Status

~~This EAL is for PWRs using Critical Safety Function Status Tree (CSFST) monitoring and functional restoration procedures. For more information, please refer to Section 3.9 of this report. RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings and/or sampling results, and thus represents a potential loss of containment. Conditions leading to a containment RED path result from RCS barrier and/or Fuel Clad Barrier Loss. Thus, this EAL is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier.~~

~~There is no "Loss" EAL associated with this item.~~Not applicable

2. Containment Pressure

Rapid unexplained loss of pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure ~~increase~~rise indicates a loss of containment integrity.

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

The ~~(site-specific)~~ 55 ~~PSIG~~ psig for potential loss of containment is based on the containment design pressure. [Ref. 13, 16]

If hydrogen concentration reaches or exceeds the lower explosive limit in an oxygen rich environment, a potentially explosive mixture exists. Standard industry reference documents list the lowest potentially explosive concentration for hydrogen in a standard atmosphere as 6.0%. If the combustible mixture ignites inside containment, loss of the Containment barrier could occur. [Ref. 10] Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists. ~~The indications of potential loss under this EAL corresponds to some of those leading to the RED path in EAL #1 above and may be declared by those sites using CSFSTs. As described above, t~~This EAL is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier.

The second potential loss EAL represents a potential loss of containment in that the containment heat removal/depressurization system ~~(e.g., containment sprays, ice condenser fans, etc.,~~ (but not including containment venting strategies) are either lost or performing in a degraded manner, as indicated by containment pressure greater than the setpoint (4 psig) at which the equipment was supposed to have actuated. [Ref. 11, 17]

3. Core Exit Thermocouples

In this EAL, the ~~function~~ restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is ~~decreasing~~ lowering or if the vessel water level is ~~increasing~~rising.

~~For units using the CSF status trees a direct correlation to those status trees can be made if the effectiveness of the restoration procedures is also evaluated as stated below.~~

CETs provide an indirect indication of fuel cladding temperature by measuring the temperature of the primary coolant that leaves the core region. Although clad rupture due to high temperature is not expected for CET readings less than the 1200 degrees F, temperatures of this magnitude signal significant superheating of the primary coolant and core uncovery. [Ref. 3]

The second part of this threshold (CET in excess of 700 degrees F and RVLMS less than 11 in. above the bottom of the fuel alignment plate) indicates loss of inventory control resulting in significant core exit superheating. The highest CET temperature expected for any FSAR analysis accident is 700 degrees F and is a limiting condition to remain in an Optimal Recovery Procedure [Ref. 2].

The RVLMS can provide indication of potential core uncovery when level decreases to 11 in. above the bottom of the fuel alignment plate, elevation 614 ft. 0 in. Therefore, RVLMS reading less than this elevation, signals inadequate coolant inventory, loss of subcooling, and the occurrence of possible fuel cladding damage. [Ref. 2, 3, 19, 20]

Severe accident analyses (e.g., NUREG-1150) have concluded that ~~function restoration~~ recovery 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~~ recovery procedures to arrest the core melt sequence. Whether or not the procedures will be effective should be apparent within 15 minutes. The Emergency Director should make the declaration as soon as it is determined that the procedures have been, or will be ineffective. ~~The reactor vessel level chosen should be consistent with the emergency response guides applicable to the facility.~~

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

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

4. SG Secondary Side Release With Primary To Secondary Leakage

This "loss" EAL recognizes that SG tube leakage can represent a bypass of the containment barrier as well as a loss of the ~~RCS~~PCS barrier. The first "loss" EAL addresses the condition in which a RUPTURED steam generator is also FAULTED. This condition represents a bypass of the ~~RCS~~PCS and containment barriers. In conjunction with ~~RCS~~PCS Barrier "loss" EAL #3, this would always result in the declaration of a Site Area Emergency. A FAULTED S/G means the existence of secondary side leakage that results in an uncontrolled lowering in steam generator pressure or the steam generator being completely depressurized. A RUPTURED S/G means the existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection, (per Section 5.0 Definitions).

The second "loss" EAL addresses SG tube leaks that exceed 10 gpm in conjunction with a nonisolable release path to the environment from the affected steam generator. The threshold for establishing the nonisolable secondary side release is intended to be a prolonged release of radioactivity from the RUPTURED steam generator directly to the environment. This could be

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

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

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

5. Containment Isolation Valve Status After Containment Isolation

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

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

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

6. Significant Radioactive Inventory in Containment

The ~~(site-specific)~~20,000 R/hr reading is a value which indicates significant fuel damage well in excess of the EALs associated with both loss of Fuel Clad and loss of ~~RCS~~PCS Barriers. [Ref. 4] ~~As stated in Section 3.8, a~~A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant.

Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of

containment, such that a General Emergency declaration is warranted. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%.

Containment radiation is indicated on Containment High Range Radiation Monitors, RIA-2321 and RIA-2322. The Containment High Range Radiation Monitors high alarm at 400 R/hr. [Ref. 5]

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

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

7. Other ~~(Site-Specific)~~ Indications

There are no "Loss" or potential loss EALs associated with this item.

~~This EAL should cover other (site-specific) indications that may unambiguously indicate loss or potential loss of the containment barrier, including indications from area or ventilation monitors in containment annulus or other contiguous buildings. If site emergency operating procedures provide for venting of the containment during an emergency as a means of preventing catastrophic failure, a Loss EAL should be included for the containment barrier. This EAL should be declared as soon as such venting is imminent. Containment venting as part of recovery actions is classified in accordance with the radiological effluent ICs.~~

8. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences. [As discussed in NEI 99-01 rev. 4]

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

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

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to EOP-9.0, "Functional Recovery." The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations. [Ref. 7, 8]

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

PNP Basis Reference(s):

1. PNP SAMG, "Phase 1.0 Initial Diagnosis"
2. EOP Setpoint Basis
3. GAMG, "Combustion Engineering Generic Accident Management Guidelines
4. NMC calculation EA-JLV-04-01, "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4"
5. ARP–33, "Auxiliary Systems Scheme EK-02 (C-11A)" Annunciator Nos. 1 and 2
6. EI-7.0, "Emergency Post Accident Sampling and Determination of Fuel Failure Using Dose Rates" Attachment 2, NSSS Sample Panel Piping Dose Rates
7. EOP-9.0, "Functional Recovery Procedure"
8. EOP-9.0, "Functional Recovery Procedure Basis"
9. ONP-23.1, "Primary Coolant Leak"
10. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident" dated November 1978
11. EOP-1.0, "Standard Post-Trip Actions"
12. FSAR Section 7.6, "CET System"
13. FSAR Section 5.8.1, "Containment Structure"
14. FSAR Section 9.10, "Chemical and Volume Control System"
15. EOP-4.0, "Loss of Coolant Accident Recovery Basis" Section 1.0 Introduction
16. FSAR Figure 14.18.1-1, "LOCA Containment Profile"
17. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13" Annunciator Nos. 40, 41, 42, 61
18. Vendor Drawing M1-BM Sh. 28, "RLI Display Panel"
19. FSAR Section 7.4.6.3 "Reactor Vessel Level Monitoring System RVLMS"
20. Vendor Drawing M-398 Sh. 1005 "Level Setting Diagram Reactor Vessel Level Monitoring System"
21. FSAR Section 7.4, "Instrumentation and Controls"

TABLE 5-H-10

Recognition Category H

Hazards and Other Conditions Affecting Plant Safety

INITIATING CONDITION MATRIX

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

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HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU1

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Natural and Destructive Phenomena Affecting the PROTECTED AREA.

Operating Mode Applicability: All

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

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

Report by Plant Personnel to on duty Control Room Personnel
AND
Confirmation from the National Earthquake Information Center

HU1.2. Report by plant personnel of tornado or high winds ~~greater than~~GREATER THAN 95~~(site-specific)~~ mph striking within PROTECTED AREA boundary.

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

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

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

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

- Emergency Diesel Generator Rooms
- Engineered Safeguards Rooms
- Auxiliary Feedwater Pump Room
- Switchgear Room 1C
- Screen House
- Component Cooling Water Pump Room

HU1.7. High lake level: ultimate heat sink level GREATER THAN 590 ft elevation affecting the PROTECTED AREA.

OR

Low lake level: ultimate heat sink level LESS THAN OR EQUAL TO 572 ft elevation.

~~(Site-Specific) occurrences affecting the PROTECTED AREA.~~

Basis:

~~N~~OUE in this IC are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators. Areas identified in the EALs define the location of the event based on the potential for damage to equipment contained therein. Escalation of the event to an Alert occurs when the magnitude of the event is sufficient to result in damage to equipment contained in the specified location.

~~EAL #1~~HU1.1~~ should be developed on site-specific basis~~. Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate. Method of detection ~~can be~~is based on ~~instrumentation,~~ validationed by a reliable source~~,~~ and ~~or~~ operator assessment. As defined in the EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, a *"felt earthquake"* is:

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

Confirmation on seismic disturbances can be obtained from the National Earthquake Information Center, Denver, Colorado, 1-303-273-8500 (normal hours), or 1-303-273-8428 (off normal hours). Seismic instrumentation is available onsite for post emergency assessment of earthquakes. [Ref. 1, 2]

~~EAL #2~~HU1.2 is based on the assumption that a tornado striking (touching down) or high winds within the PROTECTED AREA [Ref 4] may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. The high wind ~~site-specific~~ value ~~in EAL#2 should be~~is based on site-specific FSAR design basis of 100 mph [Ref. 3]. However, full-scale on the instrumentation used for classification is 100 mph. A value of 95 mph was chosen as the classification threshold, as this will still be on-scale. If such damage is confirmed visually or by other in-plant indications, the event may be escalated to Alert.

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

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

~~EAL #5~~HU1.5 is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIREs and flammable gas build up are appropriately classified via HU2 and HU3. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant. This

EAL is consistent with the definition of a ~~NO~~UE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of the emergency classification is based on potential damage done by missiles generated by the failure or ~~by the radiological releases for a BWR, or~~ in conjunction with a steam generator tube rupture~~, for a PWR~~. These latter events would be classified by the radiological ICs or Fission Product Barrier ICs. [Ref. 5]

~~EAL #6~~HU1.6 addresses the effect of flooding caused by internal events such as component failures, equipment misalignment, or outage activity mishaps. The site-specific areas include those areas that contain systems required for safe shutdown of the plant, that are not designed to be wetted or submerged. Escalation of the emergency classification is based on the damage caused or by access restrictions that prevent necessary plant operations or systems monitoring. [Ref. 6, 7, 8]~~The plant's IPEEE may provide insight into areas to be considered when developing this EAL.~~

~~EAL #7~~HU1.7 covers ~~other site-specific phenomena such as hurricane,~~ external flooding and low lake level.~~, or seiche. These EALs can also be precursors of more serious events. In particular, sites subject to severe weather as defined in the NUMARC station blackout initiatives, should include an EAL based on activation of the severe weather mitigation procedures (e.g., precautionary shutdowns, diesel testing, staff call-outs, etc.).~~ The nominal lake level is approximately 580 ft elevation.

The high lake water level threshold is ground level, which is approximately four feet below the design flood-seiche event level of 594.1 ft elevation. This level is obtained by adding a 10.9 ft onshore surge height to a maximum monthly mean lake level of 583.2 ft elevation. When lake water level approaches 590 ft elevation., ONP-12, "Acts of Nature", requires initiation of a plant shutdown [Ref. 2, 7, 9].

The low lake level is 216 in. below the Screen House floor level and ensures water level is well above the sluice gate opening which is at 568.25 ft elevation. Water level at this elevation ensures adequate NPSH to SWS pumps. The NPSH calculation assumes a minimum water level of 4 feet above the bottom of the pump suction bell which corresponds to an elevation of 557.25 ft. Violation of the SWS pump submergence requirement should never become a factor unless the Lake Michigan water level falls below the top of the sluice gate opening. ONP-6.1, "Loss of Service Water", requires tripping the reactor when service water bay level lowers below 572 ft. [Ref. 10]

PNP Basis Reference(s):

1. FSAR Section 2.4, "Seismicity"
2. ONP-12, "Acts of Nature"
3. FSAR Section 5.3.1.1, "Wind and Tornado Loadings – Design Parameters"
4. FSAR Figure 1-1, "Plant Area Plan"
5. FSAR Section 5.5, "Missile Protection"
6. DBD 7.08, "Plant Protection against Flooding"
7. FSAR Section 5.4, "Water Level Design"
8. FSAR Section 5.6, "Dynamic Effects Of Pipe Rupture"
9. FSAR Section 2.2.2, "General Lake Hydrology"
10. ONP-6.1, "Loss of Service Water"

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

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

Operating Mode Applicability: All

~~Example~~ Emergency Action Level:

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

Table H-1 Plant VITAL AREAS
• Containment Structure • Auxiliary Building • Turbine Building • Screenhouse

~~(Site-specific) list~~

Basis:

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

The intent of this 15 minute duration is to size the FIRE and to discriminate against small FIREs that are readily extinguished (e.g., smoldering waste paper basket). The ~~site-specific list should be~~Table H-1 areas are limited and ~~applies~~ apply to buildings and areas contiguous (in actual contact with or immediately adjacent) to plant VITAL AREAs or other significant buildings or areas. The intent of this IC is not to include buildings (i.e., warehouses) or areas that are not contiguous (in actual contact with or immediately adjacent) to plant VITAL AREAs. This EAL excludes FIREs within non-contiguous administration buildings, waste-basket FIREs~~,~~ and other small FIREs of no safety consequence.

Escalation to a higher emergency class is by IC HA24, ~~"FIRE Affecting the Operability of Plant Safety Systems Required for the Current Operating Mode"~~ "FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown." [Ref. 1]

PNP Basis Reference(s):

1. ONP-25.1, "Fire Which Threatens Safety-Related Equipment"

HU3

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

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

Operating Mode Applicability: All

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

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

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

Basis:

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

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

PNP Basis Reference(s):
None

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU4

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

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

Operating Mode Applicability: All

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

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

HU4.1. Security Shift Leader reports ANY of the following:

- Suspected SABOTAGE device discovered within the plant PROTECTED AREA
- Suspected SABOTAGE device discovered outside the PROTECTED AREA or in the plant switchyard
- Confirmed tampering with safety-related equipment
- A HOSTAGE/EXTORTION situation that disrupts NORMAL PLANT OPERATIONS
- CIVIL DISTURBANCE or strike which disrupts NORMAL PLANT OPERATIONS
- Internal disturbance that is not short lived or that is not a harmless outburst involving ANY individuals within the PROTECTED AREA
- Malevolent use of a vehicle outside the PROTECTED AREA which disrupts NORMAL PLANT OPERATIONS

HU4.2. ~~2.~~ A credible site specific security threat notification.

Basis:

Reference is made to ~~(site-specific) security shift supervision~~the Security Shift Leaders because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security ~~Safeguards Contingency~~ Plan. [Ref. 1]

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

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

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

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

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

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

PNP Basis Reference(s):

1. Security Plan (Safeguards Information)

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU5

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a ~~N~~OUE.

Operating Mode Applicability: All

~~Example~~ Emergency Action Level:

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

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the ~~N~~OUE 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.

PNP Basis Reference(s):
None

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA1

Initiating Condition -- ALERT

Natural and Destructive Phenomena Affecting the Plant VITAL AREA.

Operating Mode Applicability: All

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

HA1.1. Seismic event GREATER THAN 0.1 g Operating Basis Earthquake (OBE) as indicated by VISIBLE DAMAGE to any of the following (Table H-2) plant structures / equipment.

Table H-2 VITAL AREAS/Safe Shutdown Equipment
• Containment Structure • Auxiliary Building • Turbine Building • Screenhouse • Safety Injection Refueling Water Tank (T-58) • Condensate Storage Tank (T-2) • Primary System Makeup Tank (T-81) • Fuel Oil Storage Tank (T-10A)

~~:(Site-Specific) [method(s) of indicatinges Seismic Event greater than Operating Basis Earthquake (OBE).]~~

HA1.2. Tornado or -high winds GREATER THAN~~greater than [(site-specific, FSAR design basis]95)~~ mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following (Table H-2) plant structures / equipment or Control Room indication of degraded performance of those systems.

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

HA1.3. Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following (Table H-2) plant structures or equipment therein or ~~c~~Control Room indication of degraded performance of those systems:.

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

HA1.4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any of the following (Table H-2) plant areas.~~:~~

~~(site-specific) list.~~

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

- Emergency Diesel Generator Rooms
- Engineered Safeguards Rooms
- Auxiliary Feedwater Pump Room
- Switchgear Room 1C
- Screen House
- Component Cooling Water Pump Room

HA1.6. ~~(Site-Specific)~~ The following occurrences within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to plant structures containing equipment necessary for safe shutdown, or has caused damage as evidenced by ~~c~~Control R~~r~~oom indication of degraded performance of those systems.

High lake level: ultimate heat sink level GREATER THAN 594 ft elevation

OR

Low lake level: ultimate heat sink level LESS THAN OR EQUAL TO 569 ft elevation.

Basis:

The Table H-2 areas contain systems and components required for the safe shutdown functions of the plant. The PNP safe shutdown analyses were consulted for equipment and plant areas required for the applicable mode. [Ref. 11] The EALs in this IC escalate from the ~~N~~OUE EALs in HU1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by C~~c~~ontrol Room indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this

degradation. Escalation to higher classifications occur on the basis of other ICs (e.g., System Malfunction).

~~EAL #1~~HA1.1 is based on the FSAR operating basis earthquake (OBE) of 0.1 g acceleration. ~~should be based on site-specific FSAR design basis.~~ Seismic events of this magnitude can result in a plant VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems. ~~See EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, for information on seismic event categories.~~
Confirmation on seismic disturbances can be obtained from the National Earthquake Information Center, Denver, Colorado, 1-303-273-8500 (normal hours), or 1-303-273-8428 (off normal hours). Seismic instrumentation is available onsite for post emergency assessment of earthquakes. [Ref.1, 2]

~~EAL #2~~HA1.2 is based on the FSAR design basis sustained wind speed of 100 mph. However, full-scale on the instrumentation used for classification is 100 mph. A value of 95 mph was chosen as the classification threshold, as this will still be on-scale.~~should be based on site-specific FSAR design basis~~ [Ref. 3] Wind loads of this magnitude can cause damage to safety functions.

EAL #s HA1.2, HA1.3, HA1.4 and HA1.5 ~~2, 3, 4, 5 should~~ specify site-specific structures or areas containing systems and functions required for safe shutdown of the plant.

~~EAL #3~~HA1.3 is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant. [Ref. 4]

~~EAL #4~~HA1.4 is intended to address the threat to safety related equipment imposed by missiles generated by main turbine rotating component failures. Table H-2 lists areas that contain systems and components required for the safe shutdown functions of the plant.~~This site-specific list of areas should include all areas containing safety-related equipment, their controls, and their power supplies~~. This EAL is, therefore, consistent with the definition of an ALERT in that if missiles have damaged or penetrated areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant. [Ref. 4, 5]

~~EAL #5~~HA1.5 addresses the effect of internal flooding that has resulted in degraded performance of systems affected by the flooding, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant. This flooding may have been caused by internal events such as component failures, equipment misalignment, or outage activity mishaps. The site-specific areas includes those areas that contain systems required for safe shutdown of the plant, that are not designed to be wetted or submerged. ~~The plant's IPEEE may provide insight into areas to be considered when developing this EAL.~~[Ref 6, 7, 8]

~~EAL #6~~HA1.6 covers ~~other site-specific phenomena such as hurricane,~~ flooding, low lake level or seiche. ~~These~~ This EAL~~s~~ can ~~also~~ be a precursor~~s~~ of more serious events. The high lake water level threshold is the design flood-seiche event level of 594.1 ft elevation. (rounded down to 594 ft). This level is obtained by adding a 10.9 ft onshore surge height to a maximum monthly mean lake level of 583.2 ft elevation. When lake water level approaches 590 ft elevation, ONP-12, "Acts of Nature", requires initiation of a plant shutdown. When lake water level exceeds 590 ft water may start to enter the Turbine Building/Screen House, ONP-12 requires tripping the reactor. [Ref. 2, 7, 9]

The low lake water level corresponds to the Screen House sluice gate opening of 568.25 ft elevation. (rounded to 569 ft). Water level at this elevation ensures adequate NPSH to SWS pumps. The NPSH calculation assumes a minimum water level of 4 feet above the bottom of the pump suction bell which corresponds to an elevation of 557.25 ft. Violation of the SWS pump submergence requirement should never become a factor unless the Lake Michigan water level falls below the top of the sluice gate opening. [Ref. 10].

PNP Basis Reference(s):

1. FSAR Section 2.4, "Seismicity"
2. ONP-12, "Acts of Nature"
3. FSAR Section 5.3.1.1, "Wind and Tornado Loadings – Design Parameters"
4. FSAR Figure 1-1, "Plant Area Plan"
5. FSAR Section 5.5, "Missile Protection"
6. DBD 7.08, "Plant Protection against Flooding"
7. FSAR Section 5.4, "Water Level Design"
8. FSAR Section 5.6, "Dynamic Effects Of Pipe Rupture"
9. FSAR Section 2.2.2, "General Lake Hydrology"
10. ONP-6.1, "Loss of Service Water"
11. ONP-25.1, "Fire Which Threatens Safety-Related Equipment"

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA2

Initiating Condition -- ALERT

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

Operating Mode Applicability: All

~~Example~~ Emergency Action Level:

HA2.1. FIRE or EXPLOSION in any of the following (Table H-2) ~~(site-specific)~~ areas~~:~~.

Table H-2 VITAL AREAS/Safe Shutdown Equipment
• Containment Structure • Auxiliary Building • Turbine Building • Screenhouse • Safety Injection Refueling Water Tank (T-58) • Condensate Storage Tank (T-2) • Primary System Makeup Tank (T-81) • Fuel Oil Storage Tank (T-10A)

~~(Site-specific) list~~

AND

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

Basis:

The Table H-2 areas contain systems and components required for the safe shutdown functions of the plant. The PNP safe shutdown analyses were consulted for equipment and plant areas required for the applicable mode. ~~Site-specific areas containing functions and systems required for the safe shutdown of the plant should be specified. Site-Specific Safe Shutdown Analysis should be consulted for equipment and plant areas required to establish or maintain safe shutdown.~~ This will make it easier to determine if the FIRE or EXPLOSION is potentially affecting one or more redundant trains of safety systems [Ref. 1]. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels / Radiological Effluent, or Emergency Director Judgment ICs.

This EAL addresses a FIRE / EXPLOSION and not the degradation in performance of affected systems. System degradation is addressed in the System Malfunction EALs. The reference to damage of systems is used to identify the magnitude of the FIRE / EXPLOSION and to discriminate against minor FIREs / EXPLOSIONs. The reference to safety systems is included to discriminate against FIREs / EXPLOSIONs in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the FIRE / EXPLOSION was large enough to cause damage to these systems. Thus, the designation of a single train was intentional and is appropriate when the FIRE / EXPLOSION is large enough to affect more than one component.

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

The inclusion of a "report of VISIBLE DAMAGE" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the EXPLOSION with reports of evidence of damage is sufficient for declaration. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency Director with the resources needed to perform these damage assessments. The Emergency Director also needs to consider any security aspects of the EXPLOSIONs, if applicable. [Ref. 1]

PNP Basis Reference(s):

1. ONP-25.1, "Fire Which Threatens Safety-Related Equipment"

Initiating Condition -- ALERT

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

Operating Mode Applicability: All

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

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

Table H-1 Plant VITAL AREAS
• Containment Structure • Auxiliary Building • Turbine Building • Screenhouse

HA3.2. Report or detection of gases in concentration greater than the LOWER FLAMMABILITY LIMIT within or contiguous to a VITAL AREA (Table H-1).

Basis:

This IC is based on gases that affect the safe operation of the plant. This IC applies to buildings and areas contiguous to plant VITAL AREAs or other significant buildings or areas ~~(i.e., service water pump house)~~ [Ref. 1]. The intent of this IC is not to include buildings (e.g., warehouses) or other areas that are not contiguous or immediately adjacent to plant VITAL AREAs. It is appropriate that increased monitoring be done to ascertain whether consequential damage has occurred. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels / Radioactive Effluent, or Emergency Director Judgment ICs.

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

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

PNP Basis Reference(s):

1. ONP-25.1, "Fire Which Threatens Safety-Related Equipment"

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA4

Initiating Condition -- ALERT

Confirmed Security Event in a Plant PROTECTED AREA.

Operating Mode Applicability: All

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

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

HA4.2. Security Shift Leader reports any of the following:

- SABOTAGE device discovered in the plant PROTECTED AREA
- Standoff attack on the PROTECTED AREA by a HOSTILE FORCE (i.e., Sniper)
- ANY Security event of increasing severity that persists for GREATER THAN 30 min.:
  - Credible BOMB threats
  - HOSTAGE/EXTORTION
  - Suspicious FIRE OR EXPLOSION
  - Significant Security System Hardware Failure
  - Loss of Guard Post Contact

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

Basis:

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

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

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

Reference is made to ~~(site-specific) s~~Security ~~s~~Shift ~~supervisor~~ Leaders because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan. [Ref. 1]

PNP Basis Reference(s):

1. Security Plan (Safeguards)

HA5

Initiating Condition -- ALERT

Control Room Evacuation Has Been Initiated.

Operating Mode Applicability: All

~~Example~~ Emergency Action Level:

HA5.1. Entry into ONP-25.2, "Alternate Safe Shutdown" ~~[(site-specific) procedure number(s) and title(s)]~~ for ~~c~~Control ~~r~~Room evacuation.

Basis:

With the ~~c~~Control ~~r~~Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facility is necessary. ONP-25.2, "Alternate Safe Shutdown", provides specific instructions for evacuating the Control Room and establishing plant control at remote Panels C-150 Auxiliary Hot Shutdown Panel/C-150A Auxiliary Hot Shutdown Monitoring Panel. Inability to establish plant control from outside the control room will escalate this event to a Site Area Emergency. [Ref. 1]

PNP Basis Reference(s):

1. ONP-25.2, "Alternate Safe Shutdown"

HA6

Initiating Condition -- ALERT

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

Operating Mode Applicability: All

~~Example~~ Emergency Action Level:

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

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency class.

PNP Basis Reference(s):
None

Initiating Condition – SITE AREA EMERGENCY

Confirmed Security Event in a Plant VITAL AREA.

Operating Mode Applicability: All

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

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

HS1.2. Security Shift Leader reports any of the following:
- SABOTAGE discovered in a VITAL AREA.
- HOSTAGE/EXTORTION in a VITAL AREA.

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

Basis:

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

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

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

Reference is made to ~~(site-specific) s~~ Security ~~shift~~ Shift ~~supervision~~ Leaders because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan. [Ref 1]

PNP Basis Reference(s):
1. Security Plan

Initiating Condition – SITE AREA EMERGENCY

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

Operating Mode Applicability: All

~~Example~~ Emergency Action Level:

HS2.1. Control ~~r~~Room evacuation has been initiated.

AND

Control of the plant cannot be established per ONP- 25.2, "Alternate Safe Shutdown" ~~(site-specific)~~ procedure within ~~(site-specific)~~ 15 minutes.

Basis:

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

ONP-25.2, Alternate Safe Shutdown, provides specific instructions for evacuating the Control Room and establishing plant control at the remote Panels C-150 Auxiliary Hot Shutdown Panel / C-150A Auxiliary Hot Shutdown Monitoring Panel. ONP-25.2 also specifies required times for completing individual operator actions to control the plant. The required times for completing all of these actions are greater than 15 minutes with the exception of isolating atmospheric steam dumps. Steam dumps need to be isolated within six minutes to avoid overcooling the PCS. However, this action would be completed prior to establishing control at C-150/C-150A. The 15 minute limit is established to ensure control is established at C-150/C-150A in sufficient time to allow completion of the remaining actions in ONP-25.2. [Ref. 1]

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

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

PNP Basis Reference(s):

1. ONP-25.2, "Alternate Safe Shutdown"

HS3

Initiating Condition – SITE AREA EMERGENCY

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

Operating Mode Applicability: All

~~Example~~ Emergency Action Level:

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

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency class description for Site Area Emergency.

PNP Basis Reference(s):
None

HG1

Initiating Condition – GENERAL EMERGENCY

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

Operating Mode Applicability: All

~~Example~~ Emergency Action Level:

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

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

OR

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

Basis:

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

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

Loss of physical control of the control room or remote shutdown capability alone may not prevent the ability to maintain safety functions per se.~~ Design of the remote shutdown capability and the location of the transfer switches should be taken into account.~~[Ref. 1]

PNP Basis Reference(s):

1. Security Plan (Safeguards)

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG2

Initiating Condition – GENERAL EMERGENCY

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

Operating Mode Applicability: All

~~Example~~ Emergency Action Level:

HG2.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the General Emergency class.

PNP Basis Reference(s):
None

Table S-0

Recognition Category S

System Malfunction

INITIATING CONDITION MATRIX

~~NOUE~~	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
SU1 Loss of All Offsite Power to Essential Busses for ~~Greater Than~~GREATER THAN 15 Minutes. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SA5** AC power capability to essential busses reduced to a single power source for ~~greater than~~ GREATER THAN 15 minutes such that any additional single failure would result in station blackout. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SS1** Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SG1** Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*
	SA2 Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram Was Successful. *Op. Modes: Power Operation, Startup, Hot Standby*	**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. *Op. Modes: Power Operation, Startup*	**SG2** Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core. *Op. Modes: Power Operation, Startup*
SU2 Inability to Reach Required Shutdown Within Technical Specification Limits. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SA3** *Deleted*	**SS4** Complete Loss of Heat Removal Capability. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	
SU3 UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for ~~Greater Than~~ GREATER THAN15 Minutes *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SA4** UNPLANNED Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a SIGNIFICANT TRANSIENT in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SS6** Inability to Monitor a SIGNIFICANT TRANSIENT in Progress. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	

Recognition Category S

System Malfunction

INITIATING CONDITION MATRIX

SU7 *Deleted*

SU4 Fuel Clad Degradation.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SU5 ~~RCS~~ PCS Leakage.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SU6 UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SU8 Inadvertent Criticality.
Op Modes: Hot Standby, Hot Shutdown

SA1 *Deleted*

SS3 Loss of All Vital DC Power.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SS5 *Deleted*

SYSTEM MALFUNCTION

SU1

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Loss of All Offsite Power to Essential Busses for ~~Greater Than~~GREATER THAN 15 Minutes.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example~~ Emergency Action Level:

SU1.1. Loss of all offsite power to both Vital 2400 VAC busses 1C and 1D ~~(site-specific) transformers~~ for GREATER THAN~~greater than~~ 15 minutes.

AND

~~At least~~Both ~~1(site-specific)~~ emergency diesel generators are supplying power to both Vital 2400 VAC ~~emergency~~ busses 1C and 1D.

Basis:

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

The 2400 VAC system consists of Safeguard Transformer 1-1, Station Power Transformer 1-2, Start Up Transformer 1-2, four 2400 VAC busses (1C, 1D, 1E and Safeguards). Busses 1C and 1D are the VITAL (essential) busses that supply power to engineered safeguards loads. [Ref. 1] Diesel Generator 1-1 will start when an undervoltage is sensed on 2400 Volt Bus 1C and Diesel Generator 1-2 will start when an undervoltage is sensed on 2400 Volt Bus 1D. [Ref. 2]

~~Plants that have the capability to cross-tie AC power from a companion unit may take credit for the redundant power source in the associated EAL for this IC. Inability to effect the cross-tie within 15 minutes warrants declaring a NOUE.~~

PNP Basis Reference(s):

1. FSAR Section 8.3.2, "Electrical Systems - 2,400 Volt System"
2. FSAR Section 8.4.1, "Electrical Systems - Emergency Generators"

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Inability to Reach Required Shutdown Within Technical Specification Limits.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example~~ Emergency Action Level:

SU2.1. Plant is not brought to required operating mode within ~~(site-specific)~~ Technical Specifications LCO Action Statement Time.

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required shutdown mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the ~~site~~ PNP Technical Specifications requires a one hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate ~~N~~OUE is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of a ~~N~~OUE is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed. Other required Technical Specification shutdowns that involve precursors to more serious events are addressed by other System Malfunction, Hazards, or Fission Product Barrier Degradation ICs.

PNP Basis Reference(s):

1. None

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for ~~Greater Than~~GREATER THAN 15 Minutes

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example~~ Emergency Action Level:

SU3.1. UNPLANNED loss of most or all ~~(site-specific)~~ annunciators or indicators associated with safety systems for ~~greater than~~GREATER THAN 15 minutes on the following:

- EC-02/12 – Reactor/PCPs
- EC-03/13 – Safety Injection/Containment Cooling & Isolation systems / Ventilation
- EC-04 – Electrical
- EC-06 – RPS
- EC-08 – Service Water/Component Cooling
- EC-11 (Rear) – Rad Monitors
- EC-11A (Front) – Control Room HVAC / Reactor Vessel Level / Core Exit Thermocouples
- EC-27 - Thermal Margin Monitors

Basis:

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer based indication equipment is considered (e.g., PPC~~SPDS, plant computer, etc.~~). [Ref. 1]

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions.

It is further recognized that ~~most~~ plant designs provides redundant safety system indication powered from separate uninterruptable power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via

10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the ~~N~~OUE is based on SU2 "Inability to Reach Required Shutdown Within Technical Specification Limits."

~~(Site-specific)~~The specified annunciators [Ref. 2, 3, 4, 5, 6, 7] or indicators [Ref. 8] for this EAL ~~must~~ include those identified in the ~~Abnormal Operating~~Off Normal Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. Due to the limited number of safety systems in operation during cold shutdown, refueling, and defueled modes, no IC is indicated during these modes of operation.

This ~~N~~OUE will be escalated to an Alert if a transient is in progress during the loss of annunciation or indication.

PNP Basis Reference(s):

1. FSAR Section 7.6.2.5, "Instrumentation and Controls - Palisades Plant Computer"
2. ARP-4, "Primary System Volume Level Pressure Scheme EK-07 (C-12)"
3. ARP-5, "Primary Coolant Pump Steam Generator and Rod Drives Scheme EK-09 (C-12)"
4. ARP-7, "Auxiliary Systems Scheme EK-11 (C-13)"
5. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (C-13)"
6. ARP-21, "Reactor Protective System Scheme EK-06 (C-06)"
7. ARP-33. "Auxiliary Systems Scheme EK-02 (C-11A)"
8. P&ID Equipment Location Reactor and Aux. Bldg. M-4 (C-5/6) [For EC-02, 03, 04, 06, 08, 11, 11A, 12, 13 & 27]

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Fuel Clad Degradation.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example~~ Emergency Action Levels: (SU4.1 or SU4.2)

SU4.1. ~~(Site-specific)~~ Any of the following radiation monitors ~~readings~~ with a VALID PPC urgent alarm indicating fuel clad degradation ~~greater than~~ GREATER THAN Technical Specification allowable limits.

Containment Isolation High Radiation Monitor - RIA-1805
Containment Isolation High Radiation Monitor - RIA-1806
Containment Isolation High Radiation Monitor - RIA-1807
Containment Isolation High Radiation Monitor - RIA-1808

SU4.2. ~~(Site-specific) coolant~~ Coolant sample activity GREATER THAN OR EQUAL TO 40 µCi/gm dose equivalent I-131 ~~value~~ indicating fuel clad degradation ~~greater than Technical Specification allowable limits~~.

Basis:

This IC is included as a ~~NO~~UE because it is considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. ~~EAL #1~~ SU4.1 addresses site-specific radiation monitor readings ~~such as BWR air ejector monitors, PWR the failed fuel monitors, etc.,~~ that provide indication of fuel clad integrity. The Containment Isolation High Radiation Monitors will initiate containment isolation on 2 out of 4 coincidence logic at a high alarm reading of 10 R/hr [Ref. 1]. Assured effectiveness of these monitors has been verified by actual testing for the case of core damage assuming approximately 1% failed fuel without LOCA [Ref. 2]. A special test conducted in 1980 in response to NUREG-0737 verified that with approximately 1% failed fuel the Containment Isolation High Radiation monitors would indicate 4.5 Rem/hour [Ref. 3]. The PPC urgent alarm is set at one tenth of that value, 0.45 Rem/hour, indicating fuel clad degradation [Ref. 4].

~~EAL #2~~ SU4.2 addresses coolant samples exceeding coolant technical specifications for iodine spike. The Technical Specification safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the 10 CFR 100 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit of 1.0 µCi/gm is permissible, if the activity levels do not exceed the limit of 40 µCi/gm for more than 48 hours.

This is acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of a SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits. [Ref. 5, 6]

Escalation of this IC to the Alert level is via the Fission Product Barrier Degradation Monitoring ICs. Though the referenced Technical Specification limits are mode dependent, it is appropriate that the EAL's be applicable in all modes, as they indicate a potential degradation in the level of safety of the plant. The companion IC to SU4 for the Cold Shutdown/Refueling modes is CU5.

PNP Basis Reference(s):

1. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (EC-13)" (Window #63)
2. FSAR Chapter 7, "Instrumentation and Controls", Section 7.3, Engineered Safeguards Controls, Step 7.3.3.3, "Design Analysis"
3. Consumers Power Company, Response to NUREG-0737, December 19, 1980 (Item II.E.4.3 - Special Test of April 15, 1980)
4. PPC Containment Isolation High Radiation Monitor Urgent Alarm Setpoints
5. Technical Specifications 3.4.16, "PCS Specific Activity"
6. Technical Specifications B 3.4.16, "PCS Specific Activity"

SYSTEM MALFUNCTION

SU5

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

~~RCS~~ PCS Leakage.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example~~ Emergency Action Levels: (SU5.1 or SU5.2)

SU5.1. Unidentified or pressure boundary leakage ~~greater than~~GREATER THAN 10 gpm.

SU5.2. Identified leakage ~~greater than~~GREATER THAN 25 gpm.

Basis:

This IC is included as a ~~NO~~UE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is observable with normal control room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. In either case, escalation of this IC to the Alert level is via Fission Product Barrier Degradation ICs.

PNP Basis Reference(s):

None

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example~~ Emergency Action Levels: (SU6.1 or SU6.2)

SU6.1. Loss of all Table C-1~~(site-specific list)~~ onsite communications capability affecting the ability to perform routine operations.

Table C-1 Onsite Communications Systems
• Telephone system
• Onsite/offsite radio system
• Public address system

SU6.2. Loss of all Table C-2~~(site-specific list)~~ offsite communications capability.

Table C-2 Offsite Communications Systems
• Telephone system
• Power failure phones
• FTS phone system
• Satellite phone

Basis:

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

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

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

~~Site-specific list for~~Table C-2 offsite communications loss [Ref. 1, 2, 3] ~~must~~ encompasses the loss of all means of communications with offsite authorities. This ~~should~~ includes the ENS, commercial telephone lines, telecopy transmissions, and dedicated phone systems.

PNP Basis Reference(s):

1. FSAR Section 7.7.8, "Instrumentation and Controls - In-Plant Communication System"
2. ~~FSAR Section 7.7.8~~SOP-31, "Plant Lighting and Communications"
3. EI-15.2, "Communications Tests"

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

Inadvertent Criticality.

~~OPERATING MODE APPLICABILITY~~Operating Mode Applicability Hot Standby

Hot Shutdown

~~Example~~ Emergency Action Level: (SU8.1~~ or SU8.2~~)

~~SU8.1. An UNPLANNED extended positive period observed on nuclear instrumentation~~
SU8.~~2~~1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

This IC addresses inadvertent criticality events. While the primary concern of this IC is criticality events that occur in Cold Shutdown or Refueling modes (NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States), the IC is applicable in other modes in which inadvertent criticalities are possible. This IC indicates a potential degradation of the level of safety of the plant, warranting an ~~NO~~UE classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated). The Cold Shutdown/Refueling IC is CU8.

This condition can be identified using the ~~period monitors/~~startup rate monitor. The term "~~extended~~sustained" is used in order to allow exclusion of expected short term positive ~~periods/~~startup rates from planned control rod movements ~~for PWRs and BWRs (~~such as shutdown bank withdrawal~~ for PWRs)~~. These short term positive ~~periods/~~startup rates are the result of the ~~increase~~ rise in neutron population due to subcritical multiplication.

Escalation would be by the Fission Product Barrier Matrix, as appropriate to the operating mode at the time of the event, or by Emergency Director Judgment.

PNP Basis Reference(s):

None

SYSTEM MALFUNCTION

SA2

Initiating Condition -- ALERT

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 Successful.

Operating Mode Applicability: Power Operation
Startup
Hot Standby

~~Example~~ Emergency Action Level:

SA2.1. ~~Indication(s) exist that indicate that reactor protection system setpoint was exceeded and automatic scram did not occur, and a successful manual scram occurred.~~
Indication(s) exist that a Reactor Protection System setpoint was exceeded

AND

RPS automatic trip did NOT occur

AND

A successful manual trip occurred from:

- EC-02 Reactor Trip pushbutton

  OR

- EC-06 Reactor Trip pushbutton

  OR

- Reactor tripped from an ATWS trip signal

Basis:

This condition indicates failure of the automatic protection system to ~~scram~~ trip the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or ~~RCS~~PCS. Reactor protection system setpoint being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because failure of the automatic protection system is the issue. A manual scram is any set of actions by the reactor operator(s) at the reactor control console which causes control rods to be rapidly inserted into the core and brings the reactor subcritical (e.g., reactor trip button, Alternate Rod Insertion). Failure of manual scram would escalate the event to a Site Area Emergency.

EOP-1.0, "Standard Post Trip Actions" instructs the operator to push both reactor trip pushbuttons on Control Room panels EC-02 and EC-06. Manual trips are not considered successful if action taken away from the Control Room panels EC-02 and EC-06 is required to trip the reactor. If opening Control Rod Drive clutch power feeder breakers 42-1 RPS and 42-2 RPS with a standing reactor trip signal present, placing all CRD clutch power toggle switches to CLUTCH OFF, or emergency borating are required to shutdown the reactor, manual trips are not considered to be successful and classification under EAL SS2.1 or SG2.1 would be required [Ref. 1, 2]. If a manual reactor trip is required to achieve reactor shutdown, a condition that is more than a potential degradation of a safety system has occurred in that a front line automatic protection system (RPS) did not function in response to a plant transient. Plant safety has thus been compromised and design limits of the fuel may have been exceeded. An Alert is indicated because conditions exist that lead to potential loss of Fuel Cladding or Primary Coolant System barrier integrity. The RPS trip setpoint being exceeded, rather than a limiting safety system setpoint, is specified here because the 'first line of defense' automatic protection system has failed.

A reactor trip from the ATWS trip signal warrants an Alert classification, as it is a 'second line of defense' trip equivalent to the manual pushbutton trips. The ATWS trip circuitry uses a diverse trip methodology. This trip methodology accomplishes a reactor trip via the same circuitry as the EC-06 Reactor Trip manual pushbutton [Ref. 3]. An Alert classification is appropriate as the ATWS trip setpoint (PCS pressure greater than 2375 psia) is designed to avoid overlap with the RPS trips or PCS safety valves [Ref. 4].

PNP Basis Reference(s):

1. EOP-1.0, "Standard Post-Trip Actions"
2. EOP-1.0, "Standard Post-Trip Actions Basis"
3. FSAR Figure 7-1 - RPS Block Diagram
4. FSAR Chapter 7, Section 7.2, "Reactor Protection System"

SYSTEM MALFUNCTION

SA4

Initiating Condition -- ALERT

UNPLANNED Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a SIGNIFICANT TRANSIENT in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example~~ Emergency Action Level:

SA4.1. UNPLANNED loss of most or all ~~(site-specific)~~ annunciators or indicators associated with safety systems for ~~greater than~~GREATER THAN 15 minutes on the following:

- EC-02/12 – Reactor/PCPs
- EC-03/13 – Safety Injection/Containment Cooling & Isolation systems / Ventilation
- EC-04 – Electrical
- EC-06 – RPS
- EC-08 – Service Water/Component Cooling
- EC-11 (Rear) – Rad Monitors
- EC-11A (Front) – Control Room HVAC / Reactor Vessel Level / Core Exit Thermocouples
- EC-27 - Thermal Margin Monitors

AND

Either of the following: ~~(a or b)~~

a. A SIGNIFICANT TRANSIENT is in progress.

OR

b. Compensatory non-alarming indications are unavailable.

Basis:

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a transient. Recognition of the availability of computer based indication equipment is considered (e.g., ~~SPDS, plant computer, etc.~~PPC) [Ref. 1].

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

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift ~~Supervisor~~ Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that ~~most~~ plant design~~s~~ provides redundant safety system indication powered from separate uninterruptable power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the ~~NOUE~~UE is based on SU2 "Inability to Reach Required Shutdown Within Technical Specification Limits."

~~Site-specific~~The specified annunciators [Ref. 2, 3, 4, 5, 6, 7] or indicators [Ref. 8] for this EAL ~~must~~ include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

"Compensatory non-alarming indications" include the PPC, plant recorders, or plant instrument displays in the control room. If both a major portion of the annunciation system and all computer monitoring are unavailable, the Alert is required. ~~Compensatory non-alarming indications" in this context includes computer-based information such as SPDS. This should include all computer systems available for this use depending on specific plant design and subsequent retrofits. If both a major portion of the annunciation system and all computer monitoring are unavailable, the Alert is required.~~

Due to the limited number of safety systems in operation during cold shutdown, refueling and defueled modes, no IC is indicated during these modes of operation.

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress.

PNP Basis Reference(s):

1. FSAR Section 7.6.2.5, "Instrumentation and Controls - Palisades Plant Computer"
2. ARP-4, "Primary System Volume Level Pressure Scheme E~~L~~K-07 (C-12)"
3. ARP-5, "Primary Coolant Pump Steam Generator and Rod Drives Scheme EK-09 (C-12)"
4. ARP-7, "Auxiliary Systems Scheme EK-11 (C-13)"
5. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (C-13)"

6. ARP-21, "Reactor Protective System Scheme EK-06 (C-06)"

7. ARP-33~~.~~, "Auxiliary Systems Scheme EK-02 (C-11A)"

8. P&ID Equipment Location Reactor and Aux. Bldg. M-4 (C-5/6) [For EC-02, 03, 04, 06, 08, 11, 11A, 12, 13 & 27]

SYSTEM MALFUNCTION

SA5

Initiating Condition -- ALERT

AC power capability to essential busses reduced to a single power source for ~~greater than~~GREATER THAN 15 minutes such that any additional single failure would result in station blackout.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example~~ Emergency Action Level:

SA5.1. AC power capability to Vital 2400 VAC ~~site specific essential buses~~busses 1C and 1D reduced to only one of the following sources~~a single power source~~ for ~~greater than~~GREATER THAN 15 minutes

- Safeguard Transformer 1-1
- Start Up Transformer 1-2
- Station Power Transformer 1-2
- 1-1 Emergency Diesel Generator
- 1-2 Emergency Diesel Generator

AND

Any additional single failure will result in station blackout.

Basis:

This IC and the associated EALs are intended to provide an escalation from IC SU1, "Loss of All Offsite Power To Essential Busses for Greater Than 15 Minutes." The condition indicated by this IC is the degradation of the offsite and onsite power systems such that any additional single failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of one emergency diesel generator to supply power to its emergency busses. Another related condition could be the loss of all offsite power and loss of onsite emergency diesels with only one train of emergency busses being backfed from the unit main generator, or the loss of onsite emergency diesels with only one train of emergency busses being backfed from offsite power.

The 2400 VAC system consists of Safeguard Transformer 1-1, Station Power Transformer 1-2, Start Up Transformer 1-2, four 2400 VAC busses (1C, 1D, 1E and Safeguards). Busses 1C and 1D are the VITAL (essential) busses that supply power to engineered safeguards loads. [Ref. 1] Diesel Generator 1-1 will start when an undervoltage is sensed on 2400 Volt Bus 1C and Diesel Generator 1-2 will start when an undervoltage is sensed on 2400 Volt Bus 1D. [Ref. 2]

The subsequent loss of ~~this~~ the single remaining power source would escalate the event to a Site Area Emergency in accordance with IC SS1, "Loss of All Offsite and Loss of All Onsite AC Power to Essential Busses."

PNP Basis Reference(s):

1. FSAR Section 8.3.2, "Electrical Systems - 2,400 Volt System"

2. FSAR Section 8.4.1, "Electrical Systems - Emergency Generators"

~~At multi-unit stations, the EALs should allow credit for operation of installed design features, such as cross-ties or swing diesels, provided that abnormal or emergency operating procedures address their use. However, these stations must also consider the impact of this condition on other shared safety functions in developing the site specific EAL.~~

SYSTEM MALFUNCTION

SS1

Initiating Condition -- SITE AREA EMERGENCY

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

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example~~ Emergency Action Level:

SS1.1. Loss of all offsite power to Vital 2400 VAC ~~buses~~busses 1C and 1D ~~power to (site-specific) transformers.~~

AND

Failure of ~~all~~both ~~(site-specific)~~ emergency diesel generators to supply power to Vital 2400 VAC busses ~~emergency busses~~.

AND

Failure to restore power to at least one Vital 2400 VAC ~~emergency~~ bus within 15 ~~(site-specific)~~ minutes from the time of loss of both offsite and onsite AC power.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including ~~RHR~~Shutdown Cooling, ~~ECCS~~Safety Injection, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will cause core uncovering and loss of containment integrity, thus this event can escalate to a General Emergency. The ~~(site-specific)~~ time duration ~~should be~~was selected to exclude transient or momentary power losses~~, but should not exceed 15 minutes~~.

The 2400 VAC system consists of Safeguard Transformer 1-1, Station Power Transformer 1-2, Start Up Transformer 1-2, four 2400 VAC busses (1C, 1D, 1E and Safeguards). Busses 1C and 1D are the VITAL (essential) busses that supply power to engineered safeguards loads. [Ref. 1] Diesel Generator 1-1 will start when an undervoltage is sensed on 2400 Volt Bus 1C and Diesel Generator 1-2 will start when an undervoltage is sensed on 2400 Volt Bus 1D. [Ref. 2]

Escalation to General Emergency is via Fission Product Barrier Degradation or IC SG1, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power."

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of AC power to Vital 2400 VAC ~~essential~~ busses. Even though ~~an~~ Vital 2400 VAC ~~essential~~ bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus then the bus should not be considered operable. If this bus was the only energized bus then a Site Area Emergency per SS1 should be declared.

PNP Basis Reference(s):

1. FSAR Section 8.3.2, "Electrical Systems - 2,400 Volt System"
2. FSAR Section 8.4.1, "Electrical Systems - Emergency Generators"

Initiating Condition -- SITE AREA EMERGENCY

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.

Operating Mode Applicability: Power Operation
Startup

~~Example~~ Emergency Action Level:

SS2.1. Indication(s) exist that automatic and manual ~~scram~~ trip were ~~not~~NOT successful.

Basis:

Automatic and manual ~~scram~~ trip are not considered successful if action away from the reactor control console was required to ~~scram~~ trip the reactor.

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency is indicated because conditions exist that lead to imminent loss or potential loss of both fuel clad and ~~RCS~~PCS. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response. Escalation of this event to a General Emergency would be via Fission Product Barrier Degradation or Emergency Director Judgment ICs.

Manual trips are not considered successful if action taken away from the Control Room panels EC-02 and EC-06 is required to trip the reactor. If opening Control Rod Drive clutch power feeder breakers 42-1 RPS and 42-2 RPS with a standing reactor trip signal present, placing all CRD clutch power toggle switches to CLUTCH OFF, or emergency borating are required to shutdown the reactor, manual trips are not considered to be successful [Ref. 1, 2]. If these actions prove unsuccessful in shutting down the reactor, escalation via EAL SG2.1 would be required. Negative startup rate and lowering reactor power are indications of successful reactor shutdown and should be observed following any reactor trip from power.

PNP Basis Reference(s):

1. EOP-1.0, "Standard Post Trip Actions"
2. EOP-1.0, "Standard Post Trip Actions Basis"

SYSTEM MALFUNCTION

SS3

Initiating Condition -- SITE AREA EMERGENCY

Loss of All Vital DC Power.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example~~ Emergency Action Level:

SS3.1. Loss of aAll v~~V~~ital DC p~~P~~ower based on ~~(site specific) bus voltage indications~~ LESS THAN 105 VDC on DC busses No. 1, ED-10 and No.2, ED-20 for ~~greater than~~GREATER THAN 15 minutes.

Basis:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. Escalation to a General Emergency would occur by Abnormal Rad Levels/Radiological Effluent, Fission Product Barrier Degradation, or Emergency Director Judgment ICs. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Station Battery #1 and Station Battery #2.have ample capacity to supply required DC loads (DC busses No. 1, ED-10 and No.2, ED-20) and preferred AC loads during a complete loss of AC power for at least four hours, assuming neither emergency diesel generator is available. The batteries are designed to furnish their maximum load down to an operating temperature of 70 degrees F without dropping below 105 VDC, and the equipment supplied by the batteries is capable of operating satisfactorily at this voltage rating. 105 VDC represents ~80% of the manufacturers' rating for battery capacity of a nominal 131 VDC. [Ref. 1]

PNP Basis Reference(s):

1. FSAR Section 8.4.2, "Electrical Systems - Station Batteries"

SYSTEM MALFUNCTION

SS4

Initiating Condition -- SITE AREA EMERGENCY

Complete Loss of Heat Removal Capability.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example~~ Emergency Action Level:

SS4.1. Loss of core cooling and heat sink ~~(PWR)~~.

~~SS4.1. Heat Capacity Temperature Limit Curve exceeded (BWR).~~

Basis:

This EAL addresses complete loss of functions, including ultimate heat sink, required for hot shutdown with the reactor at pressure and temperature. Reactivity control is addressed in other EALs. ~~For BWRs the loss of heat removal function is indicated by the Heat Removal Capability Temperature Limit Curve being exceeded.~~

Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted. Escalation to General Emergency would be via Abnormal Rad Levels / Radiological Effluent, Emergency Director Judgment, or Fission Product Barrier Degradation ICs.

PNP Basis Reference(s):

None

SYSTEM MALFUNCTION

SS6

Initiating Condition -- SITE AREA EMERGENCY

Inability to Monitor a SIGNIFICANT TRANSIENT in Progress.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example~~ Emergency Action Level:

SS6.1. ~~a.~~ Loss of most or all ~~(site-specific)~~ annunciators associated with safety systems on the following:

- EC-02/12 – Reactor/PCPs
- EC-03/13 – Safety Injection/Containment Cooling & Isolation systems / Ventilation
- EC-04 – Electrical
- EC-06 – RPS
- EC-08 – Service Water/Component Cooling
- EC-11 (Rear) – Rad Monitors
- EC-11A (Front) – Control Room HVAC / Reactor Vessel Level / Core Exit Thermocouples
- EC-27 - Thermal Margin Monitors

AND

SIGNIFICANT TRANSIENT in progress.

AND

~~b.~~ Compensatory non-alarming indications are unavailable.

AND

~~c.~~ Indications needed to monitor the ability to shut down the reactor, maintain the core cooled, maintain the reactor coolant system intact, and maintain containment intact~~monitor (site-specific) safety functions~~ are unavailable.

~~AND~~

~~d. SIGNIFICANT TRANSIENT in progress.~~

Basis:

This IC and its associated EAL are intended to recognize the inability of the control room staff to monitor the plant response to a transient. A Site Area Emergency is considered to exist if the control room staff cannot monitor safety functions needed for protection of the public.

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

"Compensatory non-alarming indications" include the PPC [Ref. 1], plant recorders, or plant instrument displays in the control room. [Ref. 8]
~~(Site-specific) annunciators for this EAL should be limited to include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., rad monitors, etc.)~~

~~"Compensatory non-alarming indications" in this context includes computer based information such as SPDS. This should include all computer systems available for this use depending on specific plant design and subsequent retrofits.~~

~~(Site-specific)~~ Indications needed to monitor safety functions necessary for protection of the public ~~must~~ include control room indications, computer generated indications and dedicated annunciation capability [Ref. 2, 3, 4, 5, 6, 7]. The specific indications ~~should be~~are those used to ~~determine such functions as~~monitor the ability to shut down the reactor, maintain the core cooled, to maintain the reactor coolant system intact, and to maintain containment intact.

"Planned" and "UNPLANNED" actions are not differentiated since the loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift ~~Supervisor~~ Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

PNP Basis Reference(s):

1. FSAR Section 7.6.2.5, "Instrumentation and Controls - Palisades Plant Computer"
2. ARP-4, "Primary System Volume Level Pressure Scheme ELK-07 (C-12)"
3. ARP-5, "Primary Coolant Pump Steam Generator and Rod Drives Scheme EK-09 (C-12)"
4. ARP-7, "Auxiliary Systems Scheme EK-11 (C-13)"
5. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (C-13)
6. ARP-21, "Reactor Protective System Scheme EK-06 (C-06)"
7. ARP-33, "Auxiliary Systems Scheme EK-02 (C-11A)"

8. P&ID Equipment Location Reactor and Aux. Bldg. M-4 (C-5/6) [For EC-02, 03, 04, 06, 08, 11, 11A, 12, 13 & 27]

SYSTEM MALFUNCTION

SG1

Initiating Condition -- GENERAL EMERGENCY

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

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

~~Example~~ Emergency Action Level:

SG1.1. Loss of all offsite power to ~~(site-specific) transformers~~Vital 2400 VAC busses 1C and 1D

AND

Failure of ~~(site-specific)~~both emergency diesel generators to supply power to ~~emergency~~ Vital 2400 VAC busses.

AND

Either of the following: ~~(a or b)~~

a. Restoration of at least one ~~emergency~~ Vital 2400 VAC bus within ~~(site-specific)~~4 hours is ~~not~~ NOT likely

OR

b. ~~(Site-Specific) Indication of c~~Continuing degradation of core cooling based on Fission Product Barrier monitoring as indicated by Average of qualified CETs GREATER THAN 700 degrees F.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including ~~RHR~~Shutdown Cooling, ~~ECCS~~Safety Injection, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will lead to loss of fuel clad, ~~RCS~~PCS, and containment. The ~~(site-specific)~~4 hours to restore AC power ~~can be~~is based on ~~a~~ the site blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout," ~~as available~~ [Ref 1]. Appropriate allowance for offsite emergency response including evacuation of surrounding areas should be considered. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.

This IC is specified to assure that in the unlikely event of a prolonged station blackout, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory.

The 2400 VAC system consists of Safeguard Transformer 1-1, Station Power Transformer 1-2, Start Up Transformer 1-2, four 2400 VAC busses (1C, 1D, 1E and Safeguards). Busses 1C and 1D are the VITAL (essential) busses that supply power to engineered safeguards loads. [Ref. 2] Diesel Generator 1-1 will start when an undervoltage is sensed on 2400 Volt Bus 1C and Diesel Generator 1-2 will start when an undervoltage is sensed on 2400 Volt Bus 1D. [Ref. 3]

The likelihood of restoring at least one emergency bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions. In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly (s)he may need to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is imminent? (Refer to Table ~~s 3 and 4~~F-1 for more information.) This threshold (CETs in excess of 700 degrees F) indicates loss of inventory control resulting in significant core exit superheating. The highest CET temperature expected for any FSAR analysis accident is 700 degrees F and is a limiting condition to remain in an Optimal Recovery Procedure [Ref. 4].

2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.

PNP Basis Reference(s):

1. FSAR Section 8.4.2, "Electrical Systems - Station Batteries"
2. FSAR Section 8.3.2, "Electrical Systems - 2,400 Volt System"
3. FSAR Section 8.4.1, "Electrical Systems - Emergency Generators"
4. EOP Setpoint Basis

SG2

Initiating Condition -- GENERAL EMERGENCY

Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core.

Operating Mode Applicability: Power Operation
Startup

~~Example~~ Emergency Action Level:

SG2.1. Indications exist that automatic and manual ~~scram~~ trip were not successful.

AND

Either of the following: ~~(a or b)~~

a. ~~Indication(s) exists that the core cooling is extremely challenged~~ Average of qualified CETs GREATER THAN 1200 degrees F.

OR

b. ~~Indication(s) exists that heat removal is extremely challenged~~ Core and PCS Heat Removal safety function status acceptance criteria cannot be met.

Basis:

Automatic and manual ~~scram~~ trip are not considered successful if action away from the reactor control console is required to ~~scram~~ trip the reactor.

Under the conditions of this IC and its associated EALs, the efforts to bring the reactor subcritical have been unsuccessful and, as a result, the reactor is producing more heat than the maximum decay heat load for which the safety systems were designed. Although there are capabilities away from the reactor control console, such as emergency boration ~~in PWRs, or standby liquid control in BWRs,~~ the continuing temperature rise indicates that these capabilities are not effective. This situation could be a precursor for a core melt sequence.

The combination of failure of both automatic and manual reactor 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 PCS barriers. Note that the plant operating mode changes to Hot Shutdown as soon as a successful reactor trip occurs. Since this EAL is applicable only to Power Operation and Startup modes, escalation to the General Emergency classification is not appropriate under this EAL after the reactor is successfully tripped. [Ref. 1, 2]

~~For PWRs, t~~The extreme challenge to the ability to cool the core is intended to mean that the core exit temperatures are at or approaching 1200 degrees F. The Average of the qualified Core Exit Thermocouples (CETs) provide an indirect indication of fuel cladding temperature by measuring the temperature of the primary coolant that leaves the core region. The extreme challenge to core

cooling threshold temperature of 1200 degrees F is consistent with CEOG Generic Accident Management Guidelines for Phase 1, "Initial Diagnosis" for core exit temperature. Although clad rupture due to high temperature is not expected for CET readings less than the threshold, temperatures of this magnitude signal severe superheating of the primary coolant and core uncovery. [Ref. 3]
~~or that the reactor vessel water level is below the top of active fuel. For plants using CSFSTs, this EAL equates to a Core Cooling RED condition and an entry into function restoration procedure FR-S.1. For BWRs, the extreme challenge to the ability to cool the core is intended to mean that the reactor vessel water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level as described in the EOP bases.~~

Another consideration is the inability to initially remove heat during the early stages of this sequence. ~~For PWRs,~~ If emergency feedwater flow is insufficient to remove the amount of heat required by design from at least one steam generator, an extreme challenge should be considered to exist. Core and PCS Heat Removal safety function status acceptance criteria are specified in the EOPs. [Ref. 4]~~For plants using CSFSTs, this EAL equates to a Heat Sink RED condition. For BWRs, considerations include inability to remove heat via the main condenser, or via the suppression pool or torus (e.g., due to high pool water temperature).~~

In the event either of these challenges exist at a time that the reactor has not been brought below the power associated with the safety system design ~~(typically 3 to 5% power)~~ a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier matrix declaration to permit maximum offsite intervention time.

PNP Basis Reference(s):

1. EOP-1.0, "Standard Post Trip Actions"
2. EOP-1.0, "Standard Post Trip Actions Basis"
3. CEOG Generic Accident Management Guidelines - Phase 1, "Initial Diagnosis"
4. EOP-9.0 "Functional Recovery Procedure" Basis (HR-1, HR-2 & HR-3)

ATTACHMENT 2

CLEAN TECHNICAL BASIS DOCUMENT

136 Pages Follow

Table of Contents

ACRONYMS

AC	Alternating Current
ATWS	Anticipated Transient Without Scram
CCW	Component Cooling Water
CDE	Committed Dose Equivalent
CE	Combustion Engineering
CFR	Code of Federal Regulations
CMT	Containment
CSF	Critical Safety Function
DC	Direct Current
DHR	Decay Heat Removal
DOT	Department of Transportation
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
ECL	Emergency Classification Level
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
EI	Emergency Implementing Procedure
EPRI	Electric Power Research Institute
ESF	Engineered Safeguards Feature
FEMA	Federal Emergency Management Agency
FSAR	Final Safety Analysis Report
GE	General Emergency
HPSI	High Pressure Safety Injection
IC	Initiating Condition
IDLH	Immediately Dangerous to Life and Health
IPEEE	Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI	Independent Spent Fuel Storage Installation
LCO	Limiting Condition of Operation
LER	Licensee Event Report
LFL	Lower Flammability Limit
LOCA	Loss of Coolant Accident

LPSI	Low Pressure Safety Injection
MSIV	Main Steam Isolation Valve
mRem	milliRem
Mw	Megawatt
NEI	Nuclear Energy Institute
NESP	National Environmental Studies Project
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Management and Resources Council
OBE	Operating Basis Earthquake
ODCM	Offsite Dose Calculation Manual
PCS	Primary Coolant System
PNP	Palisades Nuclear Plant
PPC	Palisades Plant Computer
PRA/PSA	Probabilistic Risk Assessment / Probabilistic Safety Assessment
PSIG	Pounds per Square Inch Gauge
PWR	Pressurized Water Reactor
R	Rem
RG	Regulatory Guide
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RVLMS	Reactor Vessel Level Monitoring System
SAE	Site Area Emergency
SG	Steam Generator
SI	Safety Injection
SRO	Senior Reactor Operator
SSE	Safe Shutdown Earthquake
SWS	Service Water System
TEDE	Total Effective Dose Equivalent
TOAF	Top of Active Fuel
TSC	Technical Support Center
UE	Notification Of Unusual Event

1.0 PURPOSE

This document provides the detailed set of Emergency Action Levels (EALs) applicable to the Palisades Nuclear Plant (PNP) and the associated Technical Bases using the EAL development methodology found in NEI 99-01 Revision 4 [Ref. 2.1].

The primary tool for determining the emergency classification level is the Emergency Action Level Matrix. Personnel responsible for implementation of EI-1, Emergency Classification and Actions [Ref. 2.2], and the EAL Matrix [Ref. 2.3] may use this document as a technical reference and an aid in EAL interpretation.

The user of the EAL Matrix may (but is not required) to consult the EAL Technical Basis Document in order to obtain additional information concerning the EALs under classification consideration.

2.0 REFERENCES

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

2.2 EI-1, "Emergency Classification and Actions"

2.3 Emergency Action Level Matrix

2.4 PNP Technical Specifications Table 1.1-1

2.5 GOP-14, "Shutdown Cooling Operations" Attachment 1 - Terms and Definitions.

2.6 FSAR Figure 1-1, Plant Area Plan

2.7 PNP Site Emergency Plan

3.0 DISCUSSION

3.1 Background

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the PNP Emergency Plan [Ref. 2.7].

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

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

- Consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur during plant shutdown conditions.

- Addressing initiating conditions and example emergency action levels that fully address conditions that may be postulated to occur at permanently Defueled Stations and Independent Spent Fuel Storage Installations.

- Simplifying the fission product barrier EAL threshold for a Site Area Emergency.

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

3.2 Key Definitions in EAL Methodology

The following definitions apply to the generic EAL methodology:

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

- Notification of Unusual Event (UE)

- Alert

- Site Area Emergency (SAE)

- General Emergency (GE)

Section 3.4 provides further discussion of the emergency classes.

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

- An IC is an emergency condition, which sets it apart from the broad class of conditions that may or may not have the potential to escalate into a radiological emergency.

- It can be a continuous, measurable function that is outside technical specifications, such as elevated PCS temperature or falling reactor coolant level (a symptom).

- It also encompasses occurrences such as FIRE (an event) or reactor coolant pipe failure (an event or a barrier breach).

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

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

3.3 Recognition Categories

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

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

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

3.4 Emergency Class Descriptions

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

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

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

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

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

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

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

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

SITE AREA EMERGENCY: Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels, which exceed EPA PAG exposure levels beyond the site boundary.

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

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

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

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

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

3.5 Operating Mode Applicability

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

1 Power Operations

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

2 Startup

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

3 Hot Standby

K_{eff} LESS THAN 0.99 average primary coolant temperature (Tave) GREATER THAN OR EQUAL TO 300°F

4 Hot Shutdown

K_{eff} LESS THAN 0.99 and average primary coolant temperature (Tave) LESS THAN 300°F and GREATER THAN 200°F with all reactor vessel head closure bolts fully tensioned

5 Cold Shutdown

K_{eff} LESS THAN 0.99 and average primary coolant temperature (Tave) LESS THAN OR EQUAL TO 200°F with all reactor vessel head closure bolts fully tensioned

6 Refuel

One or more reactor vessel head closure bolts less than fully tensioned

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

D Defueled

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

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

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

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

3.6 Fission Product Barriers

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

The primary fission product barriers are:

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

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

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

3.7 Emergency Classification Based on Fission Product Barrier Degradation

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

- Notification of Unusual Event:

  Any loss or any potential loss of Containment

- Alert:

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

- Site Area Emergency:

  Loss or potential loss of any two barriers

- General Emergency:

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

3.8 EAL Relationship to Emergency Operating Procedures (EOPs)

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

3.9 Symptom Based vs. Event Based Approach

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

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

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

3.10 Treatment of Emergency Class Upgrading

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

3.11 Emergency Class Downgrading

Another important aspect of usable EAL guidance is the consideration of what to do when the risk posed by an emergency is clearly decreasing.

It is recommended that a combination approach be taken involving recovery from General Emergencies and some Site Area Emergencies and termination from UEs, Alerts, and certain Site Area Emergencies causing no long-term plant damage. Downgrading to lower emergency classes adds notifications but may have merit under certain circumstances.

3.12 Classifying Transient Events

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

In general, observe the following guidance: Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met. For example, a momentary event, such as an ATWS or an earthquake, requires declaration even though the condition may have been resolved by the time the declaration is made.

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

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

3.13 Imminent EAL Thresholds

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

4.0 TECHNICAL BASES INFORMATION

4.1 Recognition Category Organization

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

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

4.2 Initiating Condition Structure

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

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

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

- IC Description

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

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

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

4.3 EAL Identification

The EAL identifier is the IC identifier followed by a period and sequence number (e.g., RU1.1, RU1.2, etc.).

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

5.0 DEFINITIONS

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

AFFECTING SAFE SHUTDOWN: event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable HOT or COLD SHUTDOWN condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

> Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in HOT SHUTDOWN. HOT SHUTDOWN is achievable, but COLD SHUTDOWN is not. This event is not "AFFECTING SAFE SHUTDOWN."
>
> Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in COLD SHUTDOWN. HOT SHUTDOWN is achievable, but COLD SHUTDOWN is not. This event is "AFFECTING SAFE SHUTDOWN."

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

CIVIL DISTURBANCE/STRIKE: A CIVIL DISTURBANCE may involve a group of unexpected or unauthorized individuals outside of the OWNER CONTROLLED AREA or PROTECTED AREA or a planned gathering as a part of a labor dispute (e.g., labor picket line, protest demonstration, etc.). When such disturbance poses a potential threat to plant safety or personnel, additional protective measures shall be reviewed and, where appropriate, implemented.

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

CONTAINMENT CLOSURE is a containment condition where at least one integral barrier to the release of radioactive material is being provided. CONTAINMENT CLOSURE Controls are used to track any impaired containment penetration so that at least one barrier to the release of radioactive material can be quickly established in the event of a loss of decay heat removal. [Ref. 2.5]

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

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

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

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

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

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

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

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

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

NORMAL PLANT OPERATIONS: activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or

administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONS.

OWNER CONTROLLED AREA is the area surrounding the Plant in which the reactor licensee has the authority to determine all activities including exclusion or removal of persons and property from the area during accident conditions.

PROTECTED AREA boundary is within the security isolation zone and is defined in FSAR Figure 1-1, Plant Area Plan. [Ref. 2.6]

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

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

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

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

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

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

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

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

6.0 EMERGENCY ACTION LEVEL TECHNICAL BASIS

Table R-0

Recognition Category R

Abnormal Rad Levels / Radiological Effluent

INITIATING CONDITION MATRIX

UE	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
RU1 Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Offsite Dose Calculation Manual (ODCM) Limits for 60 Minutes or Longer. *Op. Modes: All*	**RA1** Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Offsite Dose Calculation Manual (ODCM) Limits for 15 Minutes or Longer. *Op. Modes: All*	**RS1** Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the Actual or Projected Duration of the Release. *Op. Modes: All*	**RG1** Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology. *Op. Modes: All*
RU2 Unexpected Increase in Plant Radiation. *Op. Modes: All*	**RA3** 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 *Op. Modes: All*		
	RA2 Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel. *Op. Modes: All*		

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ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RU1

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Offsite Dose Calculation Manual(ODCM) Limits for 60 Minutes or Longer.

Operating Mode Applicability: All

Emergency Action Levels:: (RU1.1 or RU1.2 or RU1.3)

RU1.1. VALID reading on effluent monitor RIA-1049* "Liquid Radwaste Discharge Monitor" that exceeds two times the alarm setpoint* established by a current radioactivity discharge permit for 60 minutes or longer.
* with waste discharge not isolated

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

Stack Normal Range Monitor (RIA-2326)	3.2E+5 cpm
S/G Blowdown Monitor (RIA-0707)	2 X High Alarm*
Service Water Monitor (RIA-0833)	2 X High Alarm*
Turbine Bldg. Sumps Monitor (RIA-5211)	2 X High Alarm*

* with waste discharge not isolated

RU1.3. Confirmed sample analysis for gaseous or liquid release indicates concentrations or release rates, with a release duration of 60 minutes or longer, in excess of two times ODCM limit.

Basis:

This IC addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. PNP incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Offsite Dose Calculation Manual (ODCM) [Ref. 2]. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

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

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

RU1.1 addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed two times the ODCM limit and releases are not terminated within 60 minutes. The value given is two times the ODCM release limits. [Ref. 1, 3]. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance with the ODCM. Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

RU1.2 is also intended for effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared. The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. These monitor reading EALs have been determined using this methodology. The values given are two times the ODCM release limits. [Ref. 1, 3]

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

RU1.1 and RU1.2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the ODCM and is used in calculating the alarm setpoints. The fundamental basis of this IC is NOT a dose or dose rate, but rather the degradation in the level of safety of the plant implied by the uncontrolled release.

PNP Basis Reference(s):

1. NMC calculation EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4"
2. PNP ODCM
3. RGC 85-003, Setpoint Basis Correspondence RG Christie, June 28, 1985

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RU2

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Unexpected Rise in Plant Radiation.

Operating Mode Applicability: All

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

RU2.1. VALID indication of uncontrolled water level lowering to LESS THAN 646 ft. elevation in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water.

AND

UNPLANNED VALID Area Radiation Monitor reading rise on any of the following:

- Vent Monitor Fuel Handling Area (RIA-5712)
- Spent Fuel Pool Area Radiation Monitors (RIA-5709 or RIA-2313)
- Refueling Containment High Radiation (CHR) Monitors (RIA-2316 or RIA-2317)

RU2.2. Any UNPLANNED VALID Area Radiation Monitor reading rises by a factor of 1000 over normal* levels.

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

Basis:

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

In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR, explicit coverage of these types of events via RU2.1 is appropriate given their potential for increased doses to plant staff. Classification as an UE is warranted as a precursor to a more serious event. Indications include instrumentation such as water level and local area radiation monitors, and personnel (e.g., refueling crew) reports. Spent Fuel Pool Area Radiation Monitors (RIA-5709 and RIA-2313) and Refueling Containment High Radiation (CHR) Monitors (RIA-2316 or RIA-2317) also cover incidents involving the fuel transfer canal. [Ref. 4, 5] If available, video cameras may allow remote observation. Depending on available level instrumentation, the declaration threshold may need to be based on indications of water makeup rate or decrease in refueling water storage tank level.

While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the reading

on an area radiation monitor located on the refueling bridge may increase due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Generally, increased radiation monitor indications will need to combined with another indicator (or personnel report) of water loss. For refueling events where the water level drops below the Reactor Vessel flange classification would be via CU2. This event escalates to an Alert per IC RA2 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Matrix for events in operating modes 1-4.

The minimum allowable water level in the Spent Fuel Pool (SFP) and Refueling Cavity threshold is the Low Spent Fuel Pool water level, alarmed in the Control Room (annunciator EK-1309) at 646 ft elevation or 35 ft above the bottom of the pool. [Ref. 1, 2, 3]

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

PNP Basis Reference(s):

1. Technical Specifications 3.7.14 Spent Fuel Pool (SFP) Water Level
2. Technical Specifications 3.9.6 Refueling Cavity Water Level
3. ARP-8 Annunciator #9 Spent Fuel Pool Hi/Lo Level
4. FSAR Table 11-16, "Area Radiation Detectors"
5. FSAR Table 11-15, "Process Radiation Service And Equipment"

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RA1

Initiating Condition -- ALERT

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Offsite Dose Calculation Manual(ODCM) Limits for 15 Minutes or Longer.

Operating Mode Applicability: All

Emergency Action Levels: (RA1.1 or RA1.2 or RA1.3)

RA1.1. VALID reading on effluent monitor RIA-1049 "Liquid Radwaste Discharge Monitor" that exceeds 200 times the alarm setpoint* established by a current radioactivity discharge permit for 15 minutes or longer.
* with waste discharge not isolated

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

Stack Normal Range Monitor (RIA-2326)	1.3E+6 cpm
Stack High Range Monitor (RIA-2327)	1.5 mRem/hr
S/G Blowdown Monitor (RIA-0707)	200 X High Alarm*
Service Water Monitor (RIA-0833)	200 X High Alarm*
Turbine Bldg. Sumps Monitor (RIA-5211)	200 X High Alarm*

* with waste discharge not isolated

RA1.3. Confirmed sample analysis for gaseous or liquid release indicates concentrations or release rates, with a release duration of 15 minutes or longer, in excess of 200 times ODCM limit.

Basis:

This IC addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. PNP incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Offsite Dose Calculation Manual (ODCM). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

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

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

RA1.1 addresses radioactivity releases that for whatever reason cause effluent radiation monitor readings that exceed two hundred times the alarm setpoint established by the radioactivity discharge permit. The alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance with the ODCM. Indexing the EAL threshold to the ODCM setpoints in this manner insures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

RA1.2 is intended to address effluent or accident radiation monitors on non-routine release pathways (i.e., for which a discharge permit would not normally be prepared). The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. These monitor reading EALs have been determined using this methodology [Ref. 1, 3]. The limit for RIA-2326 and RIA-2327 were selected to maintain the "intervals between EALs for the four classifications" [Ref. NEI 99-1 AS1 basis].

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

RA1.1 and RA1.2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the ODCM and is used in calculating the alarm setpoints. The fundamental basis of this IC is NOT a dose or dose rate, but rather the degradation in the level of safety of the plant implied by the uncontrolled release.

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

PNP Basis Reference(s):

1. NMC calculation EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4"
2. PNP ODCM
3. RGC 85-003, Setpoint Basis Correspondence RG Christie, June 28, 1985

Initiating Condition – ALERT

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

Operating Mode Applicability: All

Emergency Action Levels: (RA2.1 or RA2.2)

RA2.1. A VALID alarm or reading on any of the following radiation monitors resulting from damage to irradiated fuel or loss of water level:

- Vent Monitor Fuel Handling Area RIA-5712 — 1E+4 cpm
- Spent Fuel Pool Area Radiation Monitors RIA-5709 or RIA-2313 — 15 mRem/hr
- Refueling Containment High Radiation (CHR) Monitors RIA-2316 or RIA-2317 — 80 mRem/hr above background

RA2.2. Water level less than 636 ft. 9 in. elevation for the reactor refueling cavity, spent fuel pool and fuel transfer canal that will result in irradiated fuel uncovering.

Basis:

This IC addresses specific events that have resulted, or may result, in unexpected increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent a degradation in the level of safety of the plant. These events escalate from IC RU2 in that fuel activity has been released, or is anticipated due to fuel heatup. This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage, which is discussed in IC EU1.

RA2.1 addresses radiation monitor indications [Ref. 1, 2, 3, 4] of fuel uncovery and/or fuel damage. Increased readings on ventilation monitors may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background at the monitor due to water level decrease may mask increased ventilation exhaust airborne activity and needs to be considered. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the monitor could in fact be properly responding to a known event involving transfer or relocation of a source, stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Application of these Initiating Conditions requires understanding of the actual radiological conditions present in the vicinity of the monitor. Information Notice No. 90-08, *"KR-85 Hazards from Decayed Fuel"* was considered in establishing radiation monitor EAL thresholds.

In RA2.2, indications include instrumentation such as water level and local area radiation monitors, and personnel (e.g., refueling crew) reports. If available, video cameras may allow remote observation. Declaration may need to be based on indications of water makeup rate or decrease in refueling water storage tank level. [Ref. 5]

Escalation, if appropriate, would occur via IC RS1 or RG1 or Emergency Director judgment.

PNP Basis Reference(s):

1. ARP 8, Safeguards Safety Injection and Isolation Scheme EK 9 (EC 9), Annunciator No. 66
2. RI-86E-1, Refueling Isolation Monitors Calibration - Source Test
3. FSAR Table 11-16, "Area Radiation Detectors"
4. FSAR Table 11-15, "Process Radiation Service And Equipment"
5. EA-KFK-90-01 "Fuel Submergence vs. Fuel Handling Activities"

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RA3

Initiating Condition -- ALERT

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

Operating Mode Applicability: All

Emergency Action Levels: (RA3.1 or RA3.2)

RA3.1. VALID radiation monitor readings GREATER THAN 15 mRem/hr in areas requiring continuous occupancy to maintain plant safety functions:

Control Room (RIA-2310)

OR

Central Alarm Station (RIA-2304)

RA3.2. Any VALID radiation monitor reading GREATER THAN 3 R/hr in areas requiring infrequent access to maintain plant safety functions (Table H-1).

Table H-1 PLANT VITAL AREAS
• Containment Structure • Auxiliary Building • Turbine Building • Screenhouse

Basis:

This IC addresses increased radiation levels that impede necessary access to operating stations, or other areas containing equipment that must be operated manually or that requires local monitoring, in order to maintain safe operation or perform a safe shutdown. It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant. The cause and/or magnitude of the increase in radiation levels is not a concern of this IC. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other IC may be involved. For example, a dose rate of 15 mRem/hr in the control room may be a problem in itself. However, the increase may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, an SAE or GE may be indicated by the fission product barrier matrix ICs.

This IC is not meant to apply to increases in the containment radiation monitors as these are events which are addressed in the fission product barrier matrix ICs. Nor is it intended to apply to anticipated temporary increases due to planned events (e.g., incore detector movement, radwaste container movement, depleted resin transfers, etc.)

Areas requiring continuous occupancy includes the Control Room and any other control stations that are manned continuously, such as the central security alarm station (CAS). CAS has no installed radiation monitoring capability however, RIA-2304 will provide indication of increasing radiation levels prompting surveys. [Ref. 1, 2] The value of 15mRem/hr is derived from the GDC 19 value of 5 R 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 mRem/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an Alert.

For RA3.2 areas requiring infrequent access, the basis of the 3 R/hr value is as follows:

> The PNP annual administrative personnel exposure limit is 2 R/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 R without exceeding the legal 10CFR20 annual exposure limit of 5 R and thus the need for emergency exposure authorization. Assuming that an activity required to be performed in the plant would require a 60-minute stay time in that area an area exposure rate of 3 R/hr would not unduly impede access to areas necessary for safe plant shutdown.

As used here, *impede,* includes hindering or interfering provided that the interference or delay is sufficient to significantly threaten the safe operation of the plant. Table H-1 provides the list of safe shutdown areas requiring infrequent access. The listed areas contain functions and systems required for the safe shutdown of the plant. [Ref 3]

PNP Basis Reference(s):

1. ARP 8, "Safeguards Safety Injection and Isolation Scheme EK 13 (EC 13)"
2. SOP-39, "Area Radiation Monitoring System"
3. ONP-25.1, Fire Which Threatens Safety-Related Equipment

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RS1

Initiating Condition – SITE AREA EMERGENCY

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

Operating Mode Applicability: All

Emergency Action Levels: (RS1.1 or RS1.2 or RS1.3)

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

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

Stack High Range Monitor (RIA-2327)	5.2 mRem/hr
Main Steam Line Monitors (RIA-2323/RIA-2324)	1.70E+3 cpm
Stack High Range Effluent Monitor (RIA-2328)*	0.64 mRem/hr
Atmospheric Dump Valve High Range Effluent Monitors (RIA-2328) *	0.57 mRem/hr

*Note RIA-2328 is a backup monitor which can be physically directed for monitoring the ADVs or Stack

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

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

OR

Analysis of field survey samples indicate thyroid CDE of 500 mRem for one hour of inhalation, at or beyond the site boundary.

Basis:

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

The 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 monitor list in RS1.1 includes monitors on all potential release pathways.

The EPA PAGs are expressed in terms of the sum of the *effective dose equivalent (EDE)* and the *committed effective dose equivalent (CEDE)*, or as the thyroid *committed dose equivalent (CDE).* For the purpose of these IC/EALs, the dose quantity *total effective dose equivalent (TEDE),* as defined in 10 CFR 20, is used in lieu of *"...sum of EDE and CEDE...."* The EPA PAG guidance provides for the use adult thyroid dose conversion factors.

The "SAE" effluent monitor readings are derived from Reference 1-NMC calculation, EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4."

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

PNP Basis Reference(s):

1. NMC calculation EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4"

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

RG1

Initiating Condition -- GENERAL EMERGENCY

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

Operating Mode Applicability: All

Emergency Action Levels: (RG1.1 or RG1.2 or RG1.3)

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

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

Stack high range monitor (RIA-2327)	52 mRem/hr
Main Steam Line Monitors (RIA-2323/RIA-2324)	1.70E+4 cpm
Stack high range Effluent Monitor (RIA-2328)*	6.4 mRem/hr
Atmospheric Dump Valve High Range Effluent Monitors (RIA-2328)*	5.7 mRem/hr

*Note RIA-2328 is a backup monitor which can be physically directed for monitoring the ADVs or Stack

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

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

OR

Analyses of field survey samples indicate thyroid CDE of 5000 mRem for one hour of inhalation, at or beyond site boundary.

Basis:

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

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

The monitor list in RG1.1 includes monitors on all potential release pathways.

The EPA PAGs are expressed in terms of the sum of the *effective dose equivalent (EDE)* and the *committed effective dose equivalent (CEDE)*, or as the thyroid *committed dose equivalent (CDE).* For the purpose of these IC/EALs, the dose quantity *total effective dose equivalent (TEDE),* as defined in 10 CFR 20, is used in lieu of *"...sum of EDE and CEDE...."* The EPA PAG guidance provides for the use adult thyroid dose conversion factors.

The effluent monitor readings are derived from Reference 1 - NMC calculation EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01, Revision 4."

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

PNP Basis Reference(s):

1. NMC calculation EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4"

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

INITIATING CONDITION MATRIX

UE	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
CU1 PCS Leakage. *Op. Mode: Cold Shutdown*	**CA1** Loss of PCS Inventory. *Op. Modes: Cold Shutdown*	**CS1** Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability. *Op. Modes: Cold Shutdown*	**CG1** Loss of Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the Reactor Vessel. Op. Modes: Cold Shutdown, Refueling
CU2 UNPLANNED Loss of PCS Inventory with Irradiated Fuel in the Reactor Vessel *Op. Mode: Refueling*	**CA2** Loss of Reactor Vessel Inventory with Irradiated Fuel in the Reactor Vessel. *Op. Modes: Refueling*	**CS2** Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel. Op. Modes: Refueling	
CU3 Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes. *Op. Modes: Cold Shutdown, Refueling*	**CA3** Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses. *Op. Modes: Cold Shutdown, Refueling, Defueled*		
CU4 UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel. *OP. Modes: Cold Shutdown, Refueling*	**CA4** Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel. *Op. Modes: Cold Shutdown, Refueling*		
CU5 Fuel Clad Degradation. *Op. Modes: Cold Shutdown, Refueling*			
CU6 UNPLANNED Loss of All Onsite or Offsite Communications Capabilities. *Op. Modes: Cold Shutdown, Refueling*			
CU7 UNPLANNED Loss of Required DC Power for GREATER THAN 15 Minutes. *Op. Modes: Cold Shutdown, Refueling*			
CU8 Inadvertent Criticality. *Op Modes:, Cold Shutdown, Refueling*			

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SYSTEM MALFUNCTION

CU1

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

PCS Leakage.

Operating Mode Applicability: Cold Shutdown

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

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

CU1.2. Identified leakage GREATER THAN 25 gpm.

Basis:

This IC is included as a UE because it is considered to be a potential degradation of the level of safety of the plant.

The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is sufficiently large to be observable via normally installed instrumentation or reduced inventory instrumentation such as level hose indication. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. Prolonged loss of PCS Inventory may result in escalation to the Alert level via either IC CA1 (Loss of PCS Inventory with Irradiated Fuel in the Reactor Vessel) or CA4 (Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel).

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

PNP Basis Reference(s):

None

SYSTEM MALFUNCTION

CU2

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

UNPLANNED Loss of PCS Inventory with Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: Refueling

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

CU2.1. UNPLANNED PCS level lowering below the Reactor Vessel flange for GREATER THAN OR EQUAL TO 15 minutes

CU2.2. Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump OR Primary System Drain Tank level rise

AND

Reactor Vessel level cannot be monitored

Basis:

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

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

In the refueling mode, normal means of core temperature indication and PCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of PCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing Containment Sump or Primary System Drain Tank level changes. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of PCS leakage. Rising Containment Sump level can be monitored on the PPC [Ref. 1]. Rising Primary System Drain Tank level can be monitored at EC-40, "Radwaste Panel" annunciation for which will alarm both locally and in the Control Room. [Ref. 2, 3, 4] Escalation to Alert would be via either CA2 or PCS heatup via CA4.

CU2.1 involves a decrease in PCS level below the top of the Reactor Vessel flange that continues for 15 minutes due to an UNPLANNED event. The Reactor Vessel flange is at 624 ft. 6 in. elevation and can be monitored by:

- Reactor Vessel Level indicator LIA-0105 wide range 65%
- Refueling Level Gauges LG-0105B 624 ft. 6 in. elevation. [Ref. 5]
- RVLMS UGS Region Sensor #5 uncovered (red light on): 102 in. above (fuel) bottom of fuel alignment plate or 621 ft. 8 in. elevation, which would provide additional indication that the EAL had been exceeded. [Ref. 6, 7]

This EAL is not applicable to decreases in flooded reactor cavity level (covered by RU2.1) until such time as the level decreases to the level of the vessel flange. If Reactor Vessel level continues to decrease and reaches the Bottom ID of the PCS Loop, (616 ft. 5.5 in. elevation [Ref. 5, 8]), then escalation to CA2 would be appropriate. Note that the Bottom ID of the PCS Loop Setpoint corresponds to the bottom of the Reactor Vessel loop penetration (not the low point of the loop).

PNP Basis Reference(s):

1. PPC Containment Sump Level Trends
2. P&ID Primary Coolant System M-201, Sheet 1 (A-8, G-8, F-1, D-1)
3. P&ID Radioactive Waste Treatment System Clean M-210, Sheet 2 (G-7, D-7)
4. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (C-13)" (Window #68)
5. SOP-1B, "Primary Coolant System – Cooldown"
6. VEN-M1-BM, Sheet 28 - RLI Display Panel
7. M-398, Sheet 1005 - Level Setting Diagram RVLMS8. FSAR Figure 4-2 (Hot Leg ID)

SYSTEM MALFUNCTION

CU3

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Loss of All Offsite Power to Essential Busses forGREATER THAN 15 Minutes.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Level:

CU3.1. Loss of all offsite power to Vital 2400 VAC busses 1C and 1D for GREATER THAN 15 minutes.

AND

At least one emergency diesel generator is supplying power to either Vital 2400 VAC bus 1C or 1D.

Basis:

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

The 2400 VAC system consists of Safeguard Transformer 1-1, Station Power Transformer 1-2, Start Up Transformer 1-2, four 2400 VAC busses (1C, 1D, 1E and Safeguards). Busses 1C and 1D are the VITAL (essential) busses that supply power to engineered safeguards loads. [Ref. 1] Diesel Generator 1-1 will start when an undervoltage is sensed on 2400 Volt Bus 1C and Diesel Generator 1-2 will start when an undervoltage is sensed on 2400 Volt Bus 1D. [Ref. 2]

PNP Basis Reference(s):

1. FSAR Section 8.3.2, "Electrical Systems - 2,400 Volt System"
2. FSAR Section 8.4.1, "Electrical Systems - Emergency Generators"

SYSTEM MALFUNCTION

CU4

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

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

Operating Mode Applicability: Cold Shutdown
Refueling

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

CU4.1. An UNPLANNED event results in PCS temperature exceeding the Technical Specification cold shutdown temperature limit of 200 degrees F

CU4.2. Loss of all PCS temperature and Reactor Vessel level indication for GREATER THAN 15 minutes.

Basis:

This IC is included as a UE because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or PCS inventory. Since the PCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered. In cold shutdown the decay heat available to raise PCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power. Entry into the refueling mode procedurally may not occur for typically 87 hours or longer after the reactor has been shutdown [Ref 1]. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). In addition, the operators should be able to monitor PCS temperature and Reactor Vessel level so that escalation to the alert level via CA4 or CA1 will occur if required.

During refueling the level in the Reactor Vessel will normally be maintained above the Reactor Vessel flange. Refueling evolutions that decrease water level below the Reactor Vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid rises in PCS/Reactor Vessel temperatures depending on the time since shutdown. Escalation to the Alert level is via CA4.

Unlike the cold shutdown mode, normal means of core temperature indication and PCS level indication may not be available in the refueling mode. Redundant means of Reactor Vessel level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the cold shutdown or refueling modes, CU4.2 would result in declaration of a UE if either temperature or level indication cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert would be via CA2 based on an inventory loss or CA4 based on exceeding its temperature criteria (200 degrees F) [Ref. 2].

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

PNP Basis Reference(s):

1. GOP-14, "Shutdown Cooling Operations Attachment 7"
2. Technical Specifications Table 1.1-1

SYSTEM MALFUNCTION

CU5

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Fuel Clad Degradation.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Levels: (CU5.1 or CU5.2)

CU5.1. Any of the following radiation monitors with a VALID PPC urgent alarm indicating fuel clad degradation GREATER THAN Technical Specification allowable limits.

Fuel Handling Area Monitor No. 1 - RIA-2316
Fuel Handling Area Monitor No. 2 - RIA-2317
Containment Isolation High Radiation Monitor - RIA-1805
Containment Isolation High Radiation Monitor - RIA-1806
Containment Isolation High Radiation Monitor - RIA-1807
Containment Isolation High Radiation Monitor - RIA-1808

CU5.2. Coolant sample activity GREATER THAN 1.0 µCi/gm indicating fuel clad degradation.

Basis:

This IC is included as a UE because it is considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. CU5.1 addresses site-specific radiation monitor readings that provide indication of fuel clad integrity. The Containment Isolation High Radiation Monitors will initiate containment isolation on 2 out of 4 coincidence logic at a reading of 10 R/hr [Ref. 1]. Assured effectiveness of these monitors has been verified by actual testing for the case of core damage assuming approximately 1% failed fuel without LOCA [Ref. 2]. A special test conducted in 1980 in response to NUREG-0737 verified that with approximately 1% failed fuel the Containment Isolation High Radiation monitors would indicate 4.5 Rem/hour [Ref. 3]. The PPC urgent alarm is set at one tenth of that value, 0.45 Rem/hour, indicating fuel clad degradation [Ref. 4]. The Fuel Handling Area Monitors will initiate containment isolation on 1 out of 1 coincident logic at a reading above the expected background for planned operations including movement of the reactor vessel head or internals [Ref. 5]. A calculation performed in 2004 verified that with approximately 1.0 µCi/gm activity dispersed in the flooded Reactor Cavity during refueling operations the Fuel Handling Area Radiation monitors would indicate 4.8 Rem/hour [Ref. 6]. The PPC urgent alarm is set at one tenth of that value, 0.48 Rem/hour, indicating fuel clad degradation [Ref. 7].

CU5.2 addresses coolant samples exceeding coolant technical specifications for iodine spike. The Technical Specification safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the 10 CFR 100 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit of 1.0 µCi/gm is permissible, if the activity levels do not exceed the limit of 40 µCi/gm for more than 48 hours.

This is acceptable because of the low probability of an SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

Although the Technical Specification is applicable for modes 1, 2 and 3 (when PCS Temperature GREATER THAN OR EQUAL TO 500 degrees F), it is appropriate that this EAL be applicable in cold shutdown and refueling modes, as it indicates a potential degradation in the level of safety of the plant [Ref. 8, 9].

PNP Basis Reference(s):

1. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (EC-13)" (Window 63)
2. FSAR Chapter 7, "Instrumentation and Controls", Section 7.3, Engineered Safeguards Controls, Step 7.3.3.3, "Design Analysis"
3. Consumers Power Company, Response to NUREG-0737, December 19, 1980 (Item II.E.4.3 - Special Test of April 15, 1980)
4. PPC Containment Isolation High Radiation Monitor Urgent Alarm Setpoints
5. Logic Diagram E-17, Sheet 7, "Containment High Radiation" (G-2, G-7)
6. EA-JLV-04-02R0, "Approximate Containment Radiation Monitor Response to Refueling Cavity Water Activity"
7. PPC Fuel Handling Area Radiation Monitor Urgent Alarm Setpoints
8. Technical Specifications 3.4.16, "PCS Specific Activity"
9. Technical Specifications B 3.4.16, "PCS Specific Activity" (Applicable Safety Analysis)

SYSTEM MALFUNCTION

CU6

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.

Operating Mode Applicability: Cold Shutdown
Refueling

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

CU6.1. Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations.

Table C-1 Onsite Communications Systems
• Telephone system • Onsite/offsite radio system • Public address system

CU6.2. Loss of all Table C-2 offsite communications capability.

Table C-2 Offsite Communications Systems
• Telephone system • Power failure phones • FTS phone system • Satellite phone

Basis:

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

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

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

Table C-2 offsite communications loss [Ref. 1, 2, 3] encompasses the loss of all means of communications with offsite authorities. This includes the ENS, commercial telephone lines, telecopy transmissions, and dedicated phone systems.

PNP Basis Reference(s):

1. FSAR Section 7.7.8, "Instrumentation and Controls - In-Plant Communication System"
2. SOP-31, "Plant Lighting and Communications"
3. EI-15.2, "Communications Tests"

SYSTEM MALFUNCTION

CU7

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

UNPLANNED Loss of Required DC Power forGREATER THAN 15 Minutes.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Level:

CU7.1 UNPLANNED Loss of vital DC power to required DC buses based on bus voltage indications LESS THAN 105 VDC.

AND

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

Basis:

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

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

LESS THAN 105 VDC bus voltage is based on the minimum bus voltage necessary for the operation of safety related equipment.

Station Battery #1 and Station Battery #2 have ample capacity to supply required DC loads and preferred AC loads during a complete loss of AC power for at least four hours, assuming neither emergency diesel generator is available. The batteries are designed to furnish their maximum load down to an operating temperature of 70°F without dropping below 105 VDC, and the equipment supplied by the batteries is capable of operating satisfactorily at this voltage rating. 105 VDC represents ~80% of the manufacturers' rating for battery capacity of a nominal 131 VDC. [Ref. 1]

PNP Basis Reference(s):

1. FSAR Section 8.4.2, "Electrical Systems - Station Batteries"

CU8

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Inadvertent Criticality.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Level:

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

Basis:

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

This condition can be identified using the startup rate monitor. The term "sustained" is used in order to allow exclusion of expected short term positive startup rates from planned fuel bundle or control rod movements during core alteration. These short term positive startup rates are the result of the rise in neutron population due to subcritical multiplication.

Escalation would be by Emergency Director Judgment.

PNP Basis Reference(s):

None

SYSTEM MALFUNCTION

CA1

Initiating Condition -- ALERT

Loss of PCS Inventory.

Operating Mode Applicability: Cold Shutdown

Emergency Action Levels: (CA1.1 or CA1.2)

CA1.1. Loss of PCS inventory as indicated by Reactor Vessel level LESS THAN 617 ft. 0 in. elevation.

CA1.2. Loss of PCS inventory as indicated by unexplained Containment Sump OR Primary System Drain Tank level rise

AND

Reactor Vessel level cannot be monitored for GREATER THAN 15 minutes

Basis:

These EALs serve as precursors to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further Reactor Vessel level decrease and potential core uncovery. The 616 ft. 5.5 in. elevation threshold corresponds to the bottom inside diameter of the PCS loop [Ref. 1, 2]. This condition will result in a minimum classification of Alert. The Bottom ID of the PCS Loop Setpoint was chosen because at this level remote PCS level indication may be lost and loss of suction to decay heat removal systems has occurred. The Bottom ID of the PCS Loop Setpoint is the level equal to the bottom of the Reactor Vessel loop penetration (not the low point of the loop). The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the PCS barrier.

When Reactor Vessel water level decreases to 616 ft. 5.5 in. elevation, the bottom of the PCS hot leg is uncovered. Reactor Vessel Level Transmitter LT-0105 and Reactor Hot Leg Level Transmitter LT-0106 provide the closest readily available indication of this level. Both transmitters will indicate approximately 0% at a level of 617 ft. 0 in. [Ref 2]

In cold shutdown the decay heat available to raise PCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 87 hours or longer after the reactor has been shutdown [Ref. 3]. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CA1) and a refueling specific IC (CA2).

In the cold shutdown mode, normal PCS level and Reactor Vessel level instrumentation systems will normally be available. However, if all level indication were to be lost during a loss of PCS

inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing sump and tank level changes. Rising containment sump level can be monitored on the PPC [Ref. 4]. Rising Primary System Drain Tank level can be monitored at EC-40, "Radwaste Panel" annunciation for which will alarm both locally and in the Control Room [Ref. 5, 6, 7]. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of PCS leakage. The 15-minute duration for the loss of level indication was chosen because it is half of the CS1 Site Area Emergency EAL duration. The 15-minute duration allows CA1 to be an effective precursor to CS1. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CS1 basis. Therefore this EAL meets the definition for an Alert emergency.

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

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

PNP Basis Reference(s):

1. FSAR Figure 4-2 (Hot Leg ID)
2. SOP-1B, "Primary Coolant System – Cooldown"
3. GOP-14, "Shutdown Cooling Operations" Attachment 7
4. PPC Containment Sump Level Trends
5. P&ID Primary Coolant System M-201, Sheet 1 (A-8, G-8, F-1, D-1)
6. P&ID Radioactive Waste Treatment System Clean M-210, Sheet 2 (G-7, D-7)
7. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (C-13)" (Window #68)

SYSTEM MALFUNCTION

CA2

Initiating Condition -- ALERT

Loss of Reactor Vessel Inventory with Irradiated Fuel in theReactor Vessel.

Operating Mode Applicability: Refueling.

Emergency Action Levels: (CA2.1 or CA2.2)

CA2.1. Loss of Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 617 ft. 0 in. elevation.

CA2.2. Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump OR Primary System Drain Tank level rise

AND

Reactor Vessel level cannot be monitored for GREATER THAN 15 minutes

Basis:

These EALs serve as precursors to a loss of heat removal. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further Reactor Vessel level decrease and potential core uncovery. 616 ft. 5.5 in. el. threshold corresponds to the bottom inside diameter of the PCS loop [Ref. 1, 2]. This condition will result in a minimum classification of Alert. The Bottom ID of the PCS Loop Setpoint was chosen because at this level remote PCS level indication may be lost and loss of suction to decay heat removal systems may occur. The Bottom ID of the PCS Loop Setpoint is the level equal to the bottom of the Reactor Vessel loop penetration (not the low point of the loop). The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the PCS barrier.

When Reactor Vessel water level decreases to 616 ft. 5.5 in. elevation, the bottom of the PCS hot leg is uncovered. Reactor Vessel Level Transmitter LT-0105 and Reactor Hot Leg Level Transmitter LT-0106 provide the closest reliable indication of this level. Both transmitters will indicate approximately 0% at a level of 617 ft. 0 in. [Ref 2]

In cold shutdown the decay heat available to raise PCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 87 hours or longer after the reactor has been shutdown [Ref. 3]. Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CA1) and a refueling specific IC (CA2).

In the refueling mode, normal means of Reactor Vessel level indication may not be available. Redundant means of Reactor Vessel level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of PCS inventory event, the operators would need to determine that Reactor Vessel inventory loss was occurring by observing sump and tank level changes. Rising containment sump level can be monitored on the PPC [Ref. 4]. Rising Primary System Drain Tank level can be monitored at EC-40, "Radwaste Panel" annunciation for which will alarm both locally and in the Control Room [Ref. 5, 6, 7]. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of PCS leakage. The 15-minute duration for the loss of level indication was chosen because it is half of the CS2 Site Area Emergency EAL duration. The 15-minute duration allows CA2 to be an effective precursor to CS2. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CS2 basis. Therefore this EAL meets the definition for an Alert.

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

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

PNP Basis Reference(s):

1. FSAR Figure 4-2 (Hot Leg ID)
2. SOP-1B, "Primary Coolant System – Cooldown"
3. GOP-14, "Shutdown Cooling Operations Attachment 7"
4. PPC Containment Sump Level Trends
5. P&ID Primary Coolant System M-201, Sheet 1 (A-8, G-8, F-1, D-1)
6. P&ID Radioactive Waste Treatment System Clean M-210, Sheet 2 (G-7, D-7)
7. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (C-13)" (Window #68)

SYSTEM MALFUNCTION

CA3

Initiating Condition -- ALERT

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

Operating Mode Applicability: Cold Shutdown
Refueling
Defueled

Emergency Action Level:

CA3.1. Loss of all offsite power to both Vital 2400 VAC busses 1C and 1D.

AND

Failure of both emergency diesel generators to supply power to Vital 2400 VAC busses.

AND

Failure to restore power to at least one Vital 2400 VAC bus within 15 minutes from the time of loss of both offsite and onsite AC power.

Basis:

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

The 2400 VAC system consists of Safeguard Transformer 1-1, Station Power Transformer 1-2, Start Up Transformer 1-2, four 2400 VAC busses (1C, 1D, 1E and Safeguards). Busses 1C and 1D are the VITAL (essential) busses that supply power to engineered safeguards loads. [Ref. 1] Diesel Generator 1-1 will start when an undervoltage is sensed on 2400 Volt Bus 1C and Diesel Generator 1-2 will start when an undervoltage is sensed on 2400 Volt Bus 1D. [Ref. 2]

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of AC power to essential busses. Even though an essential bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus then the bus should not be considered operable.

PNP Basis Reference(s):

1. FSAR Section 8.3.2, "Electrical Systems - 2,400 Volt System"
2. FSAR Section 8.4.1, "Electrical Systems - Emergency Generators"

SYSTEM MALFUNCTION

CA4

Initiating Condition -- ALERT

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

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Levels: (CA4.1 or CA4.2 or CA4.3)

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

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

CA4.3. An UNPLANNED event results in PCS temperature exceeding the Technical Specification cold shutdown temperature limit of 200 degrees F for GREATER THAN 60 minutes[1] OR results in a PCS pressure rise of GREATER THAN 10 psig.

Basis:

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

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

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

CA4.3 addresses complete loss of functions required for core cooling for GREATER THAN 60 minutes during refueling and cold shutdown modes when PCS integrity is established. As in CA4.1 and CA4.2, PCS integrity should be considered to be in place when the PCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). The status of CONTAINMENT CLOSURE in this EAL is immaterial given that the PCS is providing a high pressure barrier to fission product release to the environment. The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety. The 10 psig pressure rise covers situations where, due to high decay heat loads, the time provided to restore temperature control, should be less than 60 minutes. Digital PZR Pressure Indicator PI0104 narrow range (0 to 600 psia) is capable of measuring pressure to less than 25 psia. [Ref. 2] Note 1 indicates that CA4.3 is not applicable if actions are successful in restoring an PCS heat removal system to operation and PCS temperature is being reduced within the 60 minute time frame assuming that the PCS pressure rise has remained less than the site specific pressure value.

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

This IC and its associated EALs are based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." A number of phenomena such as pressurization, vortexing, steam generator U-tube draining, PCS level differences when operating at a mid-loop condition, decay heat removal system design, and level instrumentation problems can lead to conditions where decay heat removal is lost and core uncovery can occur. NRC analyses show that sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary UNPLANNED excursion above 200 degrees F [Ref. 3] when the heat removal function is available.

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

PNP Basis Reference(s):

1. GOP-14, "Shutdown Cooling Operations", Attachment 1 - Terms and Definitions
2. SOP-1B, "Primary Coolant System – Cooldown"
3. Technical Specifications Table 1.1-1

SYSTEM MALFUNCTION

CS1

Initiating Condition -- SITE AREA EMERGENCY

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

Operating Mode Applicability: Cold Shutdown

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

CS1.1. With CONTAINMENT CLOSURE not established:

a. Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 616 ft. 6 in. elevation.

OR

b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by unexplained Containment Sump OR Primary System Drain Tank level rise.

CS1.2. With CONTAINMENT CLOSURE established:

a. Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 614 ft. 0 in. elevation.

OR

b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by either:
   - Unexplained Containment Sump OR Primary System Drain Tank level rise
   - Erratic Source Range Monitor Indication

Basis:

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

In cold shutdown the decay heat available to raise PCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode. Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 87 hours or longer after the reactor has been shutdown. [Ref. 1] Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the Reactor Vessel (note that the heatup threat could be lower for cold shutdown conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CS1) and a refueling specific IC (CS2).

In the cold shutdown mode, normal PCS level and reactor vessel level monitoring systems (RVLMS) will normally be available. However, if all level indication were to be lost during a loss of PCS inventory event, the operators would need to determine that Reactor Vessel inventory loss

was occurring by observing sump and tank level changes. Rising containment sump level can be monitored on the PPC [Ref. 2]. Rising Primary System Drain Tank level can be monitored at EC-40, "Radwaste Panel" annunciation for which will alarm both locally and in the Control Room [Ref. 3, 4, 5]. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of PCS leakage. Additionally, post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors can be used as a tool for making such determinations. Two of the source range neutron flux indicators (NI-1/3A and NI-2/4A) provide control room indication [Ref. 6]. Another source range neutron flux indicator is located at Auxiliary Hot Shutdown Monitoring Panel C-150A (NI-1/3C) [Ref. 7]. Visual and audible indication is available in the Control Room and audible indication in Containment when in service. [Ref. 8].

These EALs are based on concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*, SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*, NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*, and, NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management*. A number of variables, (PWRs - e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, PCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining) can have a significant impact on heat removal capability challenging the fuel clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30-minutes was chosen.

For CS1.1 RVLMS is unable to distinguish 6" below the bottom ID of the PCS loop penetration. The closest indication of this level is provided by RVLMS UGS Region Sensor #7 red light on, ~40 in. above the fuel alignment plate at 616 ft. 6 in. el. [Ref. 9, 10, 11]. This indication allows clear escalation from CA1.1 and maintains the 6 in. difference in PCS elevation between the Alert and Site Area Emergency classification prescribed by NEI 99-01. If RVLMS is not available such that the EAL setpoint cannot be determined, then CS1.1.b should be used to determine if the IC has been met.
For CS1.2 when Reactor Vessel water level drops to 613 ft. 2 in. elevation (Top Of Active Fuel), core uncovery is about to occur. The closest indication of this level is provided by RVLMS UGS Region Sensor #8 red light on, ~11 in. above the fuel alignment plate at 614 ft. 0 in. el. [Ref. 9, 10, 11].
The 30-minute duration allowed when CONTAINMENT CLOSURE [Ref. 12] is established allows sufficient time for actions to be performed to recover needed cooling equipment and is considered to be conservative given that level is being monitored via CS1 and CS2. Effluent release is not expected with closure established.

Thus, declaration of a Site Area Emergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via CG1 (Loss of Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the Reactor Vessel) or radiological effluent IC RG1 (Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology).

PNP Basis Reference(s):

1. GOP-14, "Shutdown Cooling Operations", Attachment 7
2. PPC Containment Sump Level Trends
3. P&ID Primary Coolant System M-201, Sheet 1 (A-8, G-8, F-1, D-1)
4. P&ID Radioactive Waste Treatment System Clean M-210, Sheet 2 (G-7, D-7)
5. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (C-13)" (Window #68)

6. SOP-35, “Neutron Monitoring System”
7. EA-APR-95-001, “Appendix R Safe Shutdown Equipment List and Logic Diagrams”
8. GOP-11, “Refueling Operation and Fuel Handling”, Attachment 2 - Refuel Handling Operation Shift Checklist
9. VEN-M1-BM, Sheet 28, "RLI Display Panel"
10. M-398, Sheet 1005, "Level Setting Diagram RVLMS"
11. EOP Setpoint Basis (Top Of Active Fuel region)
12. GOP-14, “Shutdown Cooling Operations”, Attachment 1 - Terms and Definitions

SYSTEM MALFUNCTION

CS2

Initiating Condition -- SITE AREA EMERGENCY

Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capabilitywith Irradiated Fuel in the Reactor Vessel.

Operating Mode Applicability: Refueling

Emergency Action Levels: (CS2.1 or CS2.2)

CS2.1. With CONTAINMENT CLOSURE not established:

a. Reactor Vessel inventory as indicated by Rector Vessel level LESS THAN 616 ft. 6 in. elevation.

OR

b. Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by any of the following:
   - Containment High Range Radiation Monitor reading GREATER THAN 40 R/hr
   - Erratic Source Range Monitor Indication

CS2.2. With CONTAINMENT CLOSURE established

a. Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 614 ft. 0 in. elevation.

OR

b. Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by any of the following:
   - Containment High Range Radiation Monitor reading GREATER THAN 40 R/hr
   - Erratic Source Range Monitor Indication

Basis:

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

In cold shutdown the decay heat available to raise PCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode Entry into cold shutdown conditions may be attained within hours of operating at power or hours after refueling is completed. Entry into the refueling mode procedurally may not occur for typically 87 hours or longer after the reactor has been shutdown. [Ref. 1] Thus the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the refueling mode with irradiated fuel in the Reactor Vessel (note that the heatup threat could be lower for cold shutdown

conditions if the entry into cold shutdown was following a refueling). The above forms the basis for needing both a cold shutdown specific IC (CS1) and a refueling specific IC (CS2).

These EALs are based on concerns raised by Generic Letter 88-17, *Loss of Decay Heat* Removal, SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*, NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States* , and, NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.* A number of variables, – (e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, PCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining) can have a significant impact on heat removal capability challenging the fuel clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery.

For CS2.1 RVLMS is unable to distinguish 6" below the bottom ID of the PCS loop penetration. The closest indication of this level is provided by RVLMS UGS Region Sensor #7 red light on, ~40 in. above the fuel alignment plate at 616 ft. 6 in. el. [Ref. 3, 4]. This indication allows clear escalation from CA2.1 and maintains the 6 in. difference in PCS elevation between the Alert and Site Area Emergency classification prescribed by NEI 99-01.If RVLMS is not available such that the EAL setpoint cannot be determined, then CS2.1.b should be used to determine if the IC has been met.

For CS2.2 when Reactor Vessel water level drops to 613 ft. 2 in. elevation (Top Of Active Fuel) [Ref. 2], core uncovery is about to occur. The closest indication of this level is provided by RVLMS UGS Region Sensor #8 red light on, ~11 in. above the fuel alignment plate at 614 ft. 0 in. elevation. [Ref. 3, 4]

In Refueling mode, normal PCS level indication (e.g., RVLMS) 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 Containment High Range Radiation monitors RIA-2321 or RIA-2322, or erratic Source Range Monitor indication. The value of 40 Rem/hr on Containment High Range Radiation monitors RIA-2321 or RIA-2322 corresponds to the alert alarm. [Ref. 5, 6] Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and Source Range Monitors can be used as a tool for making such determinations. Two of the source range neutron flux indicators (NI-1/3A and NI-2/4A) provide control room indication [Ref. 7]. Another source range neutron flux indicator is located at Auxiliary Hot Shutdown Monitoring Panel C-150A (NI-1/3C) [Ref. 8]. Visual and audible indication is available in the Control Room and audible indication in Containment when in service. [Ref. 9]

As water level in the Reactor Vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in up-scaled Containment High Range Monitor indication and possible alarm. [Ref. 5]

For CS2.2 in the refueling mode, normal means of Reactor Vessel level indication may not be available. Redundant means of Reactor Vessel level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

Effluent release is not expected with closure established [Ref. 10].

Thus, declaration of a Site Area Emergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via CG1 (Loss of Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the Reactor Vessel) or radiological effluent IC RG1 (Offsite Dose Resulting from an Actual or Imminent Release of

Gaseous Radioactivity Exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology).

PNP Basis Reference(s):

1. GOP-14, "Shutdown Cooling Operations", Attachment 7
2. EOP Setpoint Basis (Top Of Active Fuel region)
3. VEN-M1-BM, Sheet 28, "RLI Display Panel"
4. M-398, sheet 1005, "Level Setting Diagram RVLMS"
5. NMC calculation EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4"
6. ARP 33, "Auxiliary Systems Scheme EK-02 (C-11A) Annunciator Nos. 1 and 2" (Windows #13 & #14)
7. SOP-35, "Neutron Monitoring System"
8. EA-APR-95-001, "Appendix R Safe Shutdown Equipment List and Logic Diagrams"
9. GOP-11, "Refueling Operation and Fuel Handling", Attachment 2 - Refuel Handling Operation Shift Checklist
10. GOP-14, "Shutdown Cooling Operations", Attachment 1 - Terms and Definitions

Initiating Condition -- GENERAL EMERGENCY

Loss of Reactor Vessel Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in theReactor Vessel.

Operating Mode Applicability: Cold Shutdown
Refueling

Emergency Action Level:

CG1.1. Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump OR Primary System Drain Tank level rise.

AND

Reactor Vessel Level:

a. LESS THAN 614 ft. 0 in. elevation for GREATER THAN 30 minutes

OR

b. cannot be monitored with indication of core uncovery for GREATER THAN 30 minutes as evidenced by any of the following:
   - Containment High Range Radiation Monitor reading GREATER THAN 40 R/hr
   - Erratic Source Range Monitor Indication

AND

indication of Containment challenged as indicated by any of the following:
- Containment hydrogen concentration GREATER THAN OR EQUAL TO 6%
- Containment pressure above 55 psig
- CONTAINMENT CLOSURE <u>not</u> established

Basis:

This EAL represents the inability to restore and maintain Reactor Vessel level to above the top of active fuel. Fuel damage is probable if Reactor Vessel level cannot be restored, as available decay heat will cause boiling, further reducing the Reactor Vessel level.

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, PCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining) can have a significant impact on heat removal capability challenging the fuel clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30 minutes was chosen.

Containment Sump or Primary System Drain Tank level changes may be indicative of a loss of PCS inventory. Rising containment sump level can be monitored visually on the PPC [Ref. 1]. Rising Primary System Drain Tank level can be monitored at EC-40, "Radwaste Panel" annunciation for which will alarm both locally and in the Control Room [Ref. 2, 3]. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of PCS leakage.

When Reactor Vessel water level drops to 613 ft. 2 in. elevation (Top Of Active Fuel) [Ref. 4], core uncovery is about to occur. The closest indication of this level is provided by RVLMS UGS Region Sensor #8 red light on, ~11 in. above the fuel alignment plate at 614 ft. 0 in. elevation. Fuel damage is probable if core submergence cannot be restored as available decay heat will cause boiling and further lowers the vessel level. [Ref. 5, 6]

If all means of level monitoring are not available, a Reactor Vessel inventory loss resulting in core uncovery may be detected by the following indirect methods:

- Containment High Range Radiation monitors RIA-2321 or RIA-2322 reading > 40 Rem/hr which corresponds to the alert alarm. [Ref. 7, 8]

- Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors can be used as a tool for making such determinations. Two of the source range neutron flux indicators (NI-1/3A and NI-2/4A) provide control room indication [Ref. 9]. Another source range neutron flux indicator is located at Auxiliary Hot Shutdown Monitoring Panel C-150A (NI-1/3C) [Ref. 10]. Visual and audible indication is available in the Control Room and audible indication in Containment when in service. [Ref. 11]

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

Three conditions are associated with a challenge to containment integrity:

- Standard industry reference documents list the lowest potentially explosive concentration for hydrogen in a standard atmosphere as 6.0%. [Ref. 12]

- The containment design pressure of 55 psig is in excess of that expected from the design basis loss of coolant accident. The threshold is indicative of a loss of both PCS and fuel clad boundaries in that it is not possible to reach this condition without severe core degradation. [Ref. 13]

- CONTAINMENT CLOSURE [Ref. 14] is the action taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. CONTAINMENT CLOSURE should not be confused with refueling containment integrity as defined in technical specifications. Site shutdown contingency plans typically provide for re-establishing CONTAINMENT CLOSURE following a loss of heat removal or PCS inventory functions. If the closure is re-established prior to exceeding the temperature or level thresholds of the PCS Barrier and Fuel Clad Barrier EALs, escalation to GE would not occur.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gasses in CONTAINMENT. However,

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

PNP Basis Reference(s):

1. PPC Containment Sump Level Trends
2. P&ID Primary Coolant System M-201, Sheet 1 (A-8, G-8, F-1, D-1)
3. P&ID Radioactive Waste Treatment System Clean M-210, Sheet 2 (G-7, D-7)
4. EOP Setpoint Basis (Top Of Active Fuel region)
5. VEN-M1-BM, Sheet 28, "RLI Display Panel"
6. M-398, sheet 1005, "Level Setting Diagram RVLMS"
7. NMC calculation EA-JLV-04-01 "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4"
8. ARP 33, "Auxiliary Systems Scheme EK-02 (C-11A) Annunciator Nos. 1 and 2" (Windows #13 & #14)
9. SOP-35, "Neutron Monitoring System"
10. EA-APR-95-007, "Appendix R Safe Shutdown Equipment List and Logic Diagrams"
11. GOP-11, "Refueling Operation and Fuel Handling", Attachment 2 - Refuel Handling Operation Shift Checklist
12. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident"
13. FSAR Section 5.8.1, "Design of Structures, Systems and Components - Design Basis"
14. GOP-14, "Shutdown Cooling Operations", Attachment 1 - Terms and Definitions

Table E-0

Recognition Category E

Events Related to ISFSI Malfunction

INITIATING CONDITION MATRIX

	NOUE
EU1	Damage to a loaded cask CONFINEMENT BOUNDARY. *Op. Mode: Not Applicable*
EU2	Confirmed security event with potential loss of level of safety of the ISFSI *Op. Mode: Not Applicable*

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EVENTS RELATED TO ISFSI

EU1

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

Damage to a loaded cask CONFINEMENT BOUNDARY.

Operating Mode Applicability: Not applicable

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

EU1.1. Natural phenomena events affecting a loaded cask CONFINEMENT BOUNDARY as indicated by VISIBLE DAMAGE:

- earthquake
- tornado
- flood
- seiche
- lightning

EU1.2. Accident conditions affecting a loaded cask CONFINEMENT BOUNDARY as indicated by VISIBLE DAMAGE:

- tipped over or dropped cask
- FIRE
- EXPLOSION

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

Basis:

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

Since the CONFINEMENT BOUNDARY is not directly accessible for visual inspection, the EAL definition of VISIBLE DAMAGE to the CONFINEMENT BOUNDARY is defined as: damage to the transfer cask, ventilated concrete cask, or horizontal storage module that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of the canister inside the transfer cask, ventilated concrete cask or horizontal storage module. Example damage includes: deformation due to heat, impact, or unplanned movement, denting, penetration, rupture, cracking, or spalling of concrete to expose concrete reinforcing bar, or reduction in depth or configuration of radiation shielding materials. Surface blemishes (e.g., paint fading, paint chipping, concrete cracks or scratches) are not included in VISIBLE DAMAGE.

For the events of concern here, the wind speed, earthquake intensity, height of loaded transfer cask drop as a result of normal handling or transporting etc. in and of themselves, are not the key issue. The key issue is whether the resultant damage to or loss of the loaded fuel cask CONFINEMENT BOUNDARY leads to the degradation of the fuel during transfer or storage, or poses an operational safety problem with respect to its removal from storage.

For EU1.3, any condition not explicitly detailed as an EAL threshold value, which, in the judgment of the Emergency Director, is a potential degradation in the level of safety of the ISFSI. Emergency Director judgment is to be based on known conditions and the expected response to mitigating activities within a short time period.

PNP Basis Reference(s):

1. Safety Evaluation Report For The Pacific Sierra Nuclear Associates Safety Analysis Report For The Ventilated Storage Cask System
2. "Final Safety Analysis Report For The Standardized NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel"

EVENTS RELATED TO ISFSI

EU2

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

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

Operating Mode Applicability: Not applicable

Emergency Action Levels:

EU2.1. Security Event as determined from the PNP Security Plan and reported by the Security Shift Leader.

Basis:

This EAL is based on the PNP Security Plans. Security events which do not represent a potential degradation in the level of safety of the ISFSI, are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72.

Reference is made to the Security Shift Leaders because these individuals are the designated personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Security Plan.

PNP Basis Reference(s):
1. Security Plan (Safeguards)

Table F-0

Recognition Category F

Fission Product Barrier Degradation

INITIATING CONDITION MATRIX

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

NOTES

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

   - The Fuel Clad Barrier and the PCS Barrier are weighted more heavily than the Containment Barrier. UE ICs associated with PCS and Fuel Clad Barriers are addressed under System Malfunction ICs.

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

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

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

TABLE F-1

PNP Emergency Action Level
Fission Product Barrier Reference Table
Thresholds For LOSS or POTENTIAL LOSS of Barriers*

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

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY Loss or ANY Potential Loss of Containment	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR PCS	Loss or Potential Loss of ANY Two Barriers	Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier EALS		PCS Barrier EALS		Containment Barrier EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. Critical Safety Function Status		**1. Critical Safety Function Status**		**1. Critical Safety Function Status**	
Not Applicable	Not Applicable	Not Applicable	Not Applicable	Not Applicable	Not Applicable
OR		**OR**		**OR**	
2. Primary Coolant Activity Level		**2. PCS Leak Rate**		**2. Containment Pressure**	
Dose rate value for primary coolant GREATER THAN 1 R/hr	Not Applicable	GREATER THAN available makeup capacity indicated by PCS subcooling LESS THAN 25 degrees F based on average of qualified CETs	Unisolable leak exceeding 50 gpm	Rapid unexplained lowering following initial rise OR Containment pressure or sump level response not consistent with LOCA conditions	55 psig and rising OR Containment hydrogen concentration GREATER THAN OR EQUAL TO 6% OR Containment pressure GREATER THAN 4 psig with LESS THAN one full train of depressurization equipment operating
OR				**OR**	
3. Core Exit Thermocouple Readings				**3. Core Exit Thermocouple Reading**	
GREATER THAN 1200 degrees F	GREATER THAN 700 degrees F			Not applicable	Core exit thermocouples in excess of 1200 degrees F and restoration procedures not effective within 15 minutes; - OR Core exit thermocouples in excess of 700 degrees F with reactor vessel level below 614 ft. 0 in. elevation and restoration procedures not effective within 15 minutes

TABLE F-1

PNP Emergency Action Level
Fission Product Barrier Reference Table
Thresholds For LOSS or POTENTIAL LOSS of Barriers*

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

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY Loss or ANY Potential Loss of Containment	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR PCS	Loss or Potential Loss of ANY Two Barriers	Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier EALS		PCS Barrier EALS		Containment Barrier EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
OR		**OR**		**OR**	
4. Reactor Vessel Water Level		**3. SG Tube Rupture**		**4. SG Secondary Side Release with P-to-S Leakage**	
Not Applicable	Level LESS THAN 614 ft. 0 in. elevation	SGTR that results in an ECCS (SI) Actuation	Not Applicable	RUPTURED S/G is also FAULTED outside of containment OR Primary-to-Secondary leakrate GREATER THAN 10 gpm with nonisolable steam release from affected S/G to the environment	Not applicable
				OR	
				5. CNMT Isolation Valves Status After CNMT Isolation	
				Containment Isolation Valve(s) not closed AND Downstream pathway to the environment exists after containment isolation	Not Applicable
OR		**OR**		**OR**	

TABLE F-1

PNP Emergency Action Level
Fission Product Barrier Reference Table
Thresholds For LOSS or POTENTIAL LOSS of Barriers*

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

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY Loss or ANY Potential Loss of Containment	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR PCS	Loss or Potential Loss of ANY Two Barriers	Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier

Fuel Clad Barrier EALS		PCS Barrier EALS		Containment Barrier EALS	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
5. Containment Radiation Monitoring		**4. Containment Radiation Monitoring**		**6. Significant Radioactive Inventory In Containment**	
Containment High Range Radiation Monitor reading GREATER THAN 2,000 R/hr as indicated on RIA-2321 and RIA-2322	Not Applicable	Containment High Range Radiation Monitor reading GREATER THAN 200 R/hr as indicated on RIA-2321 and RIA-2322	Not Applicable	Not Applicable	Containment High Range Radiation Monitor reading GREATER THAN 20,000 R/hr as indicated on RIA-2321 and RIA-2322
OR		**OR**		**OR**	
6. Other Indications		**5. Other Indications**		**7. Other Indications**	
Not Applicable	Not Applicable	Not Applicable	Not Applicable	Not Applicable	Not Applicable
OR		**OR**		**OR**	
7. Emergency Director Judgment		**6. Emergency Director Judgment**		**8. Emergency Director Judgment**	
Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier		Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the PCS Barrier		Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Containment barrier	

Basis Information For Table F-1
PNP Emergency Action Level
Fission Product Barrier Reference Table

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

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

1. Critical Safety Function Status

Not applicable

2. Primary Coolant Activity Level

PNP is unable to analyze a primary coolant sample of 300 uci/gm due to high dose rates experienced at the NSSS sample panel. Therefore, a dose rate is used in lieu of a sample to definitively indicate fuel clad degradation. The dose rate value used for the primary coolant is specified in accordance with Emergency Implementing Procedure, EI-7.0, "Emergency Post Accident Sampling and Determination of Fuel Failure Using Dose Rates," a sample is taken to measure contact dose rates at the Failed Fuel Survey Point on the PCS sample line. [Ref. 6] The sample is taken to measure contact dose rates at the Failed Fuel Survey Point on the PCS sample line. The site dose rate value within the first few hours for the primary coolant of approximately 1 R/hr at the NSSS sample sink is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost. [Ref. 6]

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

3. Core Exit Thermocouple Readings

Core Exit Thermocouples (CETs) provide an indirect indication of fuel cladding temperature by measuring the temperature of the primary coolant that leaves the core region. The "Loss" threshold temperature of 1200 degrees F is consistent with CEOG Generic Accident Management Guidelines, "Phase 1.0 - Initial Diagnosis," for core exit temperature. Although clad rupture due to high temperature is not expected for CET readings less than the threshold, temperatures of this magnitude signal severe superheating of the primary coolant and core uncovery. [Ref. 3]

For the "Potential Loss" threshold, CET reading significantly above the saturation temperature for the existing PCS pressure indicates a superheat condition. 700 degrees F CET temperature is therefore indicative of a potential fuel clad loss. [Ref. 2, 12] Superheat is a valid indication of a potential Fuel Cladding barrier loss condition.

4. Reactor Vessel Water Level

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

For the "Potential Loss" threshold, the RVLMS is based on a heated junction thermocouple system. The thermocouple system measures primary coolant liquid inventory with discrete sensors located at different levels within a separator tube ranging from the fuel alignment plate (i.e., near the top of active fuel) to the top of the Reactor Vessel head. The basic principle of system operation is detection of a temperature difference between heated and unheated thermocouples. [Ref. 18, 19]

Reactor Vessel water level below the top of the core may lead to a Severe Accident Management Guideline "Badly Damaged" condition. The badly damaged descriptor signifies possible core overheating to the point of clad rupture. [Ref. 3] RVLMS is the instrumentation used to indicate reactor vessel level. The lowest indication on the RVLMS is 11 inches above the bottom of the fuel alignment plate, which is approximately the top of the active fuel. RVLMS reading less than 11 inches above the bottom of the fuel alignment plate, elevation 614 ft 0 in., therefore, signals inadequate coolant inventory, loss of subcooling and the occurrence of possible fuel cladding damage. [Ref. 1, 3, 19, 20]

5. Containment Radiation Monitoring

The 2,000 R/hr reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 µCi/gm dose equivalent I-131 into the containment atmosphere. [Ref. 4] Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage. This value is higher than that specified for PCS barrier Loss EAL #4. Thus, this EAL indicates a loss of both the fuel clad barrier and a loss of PCS barrier.

Containment radiation is indicated on Containment High Range Radiation Monitors, RIA-2321 and RIA-2322. The Containment High Range Radiation Monitors high alarm at 400 R/hr. [Ref. 5]

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

6. Other Indications

There are no "Loss" or potential loss EALs associated with this item.

7. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences. [As discussed in NEI 99-01 rev. 4]

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

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

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to EOP-9.0, "Functional Recovery." The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations. [Ref. 7, 8]

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

PCS BARRIER EALs: (1 or 2 or 3 or 4 or 5 or 6)

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

1. Critical Safety Function Status

Not applicable

2. PCS Leak Rate

The "Loss" EAL addresses conditions where leakage from the PCS 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 PCS pressure and inventory against the mass loss through the leak. Subcooling margin greater than or equal to 25 degree F ensures the fluid surrounding the core is sufficiently cooled and provides margin for reestablishing flow should subcooling deteriorate when SIS flow is secured. [Ref. 2, 11, 15, 21]

The "Potential Loss" EAL is based on the inability to maintain normal liquid inventory within the Primary Coolant System (PCS) by normal operation of the Chemical and Volume Control System (CVCS). which is considered as one charging pump discharging to the charging header. [Ref. 9] The normal operating charging pump is variable speed, with a capacity of 33 to 53 gpm. The other two charging pumps are fixed speed, with a lower capacity of 40 gpm each. PNP has low capacity charging pumps, therefore a 50 gpm leak rate value is used to indicate the Potential Loss. [Ref. 2, 9,14]

3. SG Tube Rupture

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

This is consistent to the PCS Barrier "Potential Loss" EAL #2. However, if the SG is also FAULTED (i.e., two barriers failed), the declaration escalates to a Site Area Emergency per Containment Barrier "Loss" EAL #4.

There is no "Potential Loss" EAL.

4. Containment Radiation Monitoring

The 200 R/hr reading is a value which indicates the release of reactor coolant to the containment. The reading is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the containment atmosphere. [Ref. 4] This reading is less than that specified for Fuel Clad Barrier EAL #5. Thus, this EAL would be indicative of a PCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad Barrier EAL #5, fuel damage would also be indicated.

Containment radiation is indicated on Containment High Range Radiation Monitors, RIA-2321 and RIA-2322. The Containment High Range Radiation Monitors high alarm at 400 R/hr. [Ref. 5]

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

5. Other Indications

There are no "Loss" or potential loss EALs associated with this item.

6. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the PCS barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences. [As discussed in NEI 99-01 rev. 4]

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

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

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to EOP-9.0, "Functional Recovery." The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations. [Ref. 7, 8]

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

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

The Containment Barrier includes the containment building, its connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve.

1. Critical Safety Function Status

Not applicable

2. Containment Pressure

Rapid unexplained loss of pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure rise indicates a loss of containment integrity.

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

The 55 psig for potential loss of containment is based on the containment design pressure. [Ref. 13, 16]
If hydrogen concentration reaches or exceeds the lower explosive limit in an oxygen rich environment, a potentially explosive mixture exists. Standard industry reference documents list the lowest potentially explosive concentration for hydrogen in a standard atmosphere as 6.0%. If the combustible mixture ignites inside containment, loss of the Containment barrier could occur. [Ref. 10] Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists. This EAL is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier.

The second potential loss EAL represents a potential loss of containment in that the containment heat removal/depressurization system (but not including containment venting strategies) are either lost or performing in a degraded manner, as indicated by containment pressure greater than the setpoint (4 psig) at which the equipment was supposed to have actuated. [Ref. 11, 17]

3. Core Exit Thermocouples

In this EAL, the restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is lowering or if the vessel water level is rising.

CETs provide an indirect indication of fuel cladding temperature by measuring the temperature of the primary coolant that leaves the core region. Although clad rupture due to high temperature is not expected for CET readings less than the 1200 degrees F, temperatures of this magnitude signal significant superheating of the primary coolant and core uncovery. [Ref. 3]

The second part of this threshold (CET in excess of 700 degrees F and RVLMS less than 11 in. above the bottom of the fuel alignment plate) indicates loss of inventory control resulting in significant core exit superheating. The highest CET temperature expected for any FSAR analysis accident is 700 degrees F and is a limiting condition to remain in an Optimal Recovery Procedure [Ref. 2].

The RVLMS can provide indication of potential core uncovery when level decreases to 11 in. above the bottom of the fuel alignment plate, elevation 614 ft. 0 in. Therefore, RVLMS reading less than this elevation, signals inadequate coolant inventory, loss of subcooling, and the occurrence of possible fuel cladding damage. [Ref. 2, 3, 19, 20]

Severe accident analyses (e.g., NUREG-1150) have concluded that recovery 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 recovery procedures to arrest the core melt sequence. Whether or not the procedures will be effective should be apparent within 15 minutes. The Emergency Director should make the declaration as soon as it is determined that the procedures have been, or will be ineffective.

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

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

4. SG Secondary Side Release With Primary To Secondary Leakage

This "loss" EAL recognizes that SG tube leakage can represent a bypass of the containment barrier as well as a loss of the PCS barrier. The first "loss" EAL addresses the condition in which a RUPTURED steam generator is also FAULTED. This condition represents a bypass of the PCS and containment barriers. In conjunction with PCS Barrier "loss" EAL #3, this would always result in the declaration of a Site Area Emergency. A FAULTED S/G means the existence of secondary side leakage that results in an uncontrolled lowering in steam generator pressure or the steam generator being completely depressurized. A RUPTURED S/G means the existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection, (per Section 5.0 Definitions).

The second "loss" EAL addresses SG tube leaks that exceed 10 gpm in conjunction with a nonisolable release path to the environment from the affected steam generator. The threshold for establishing the nonisolable secondary side release is intended to be a prolonged release of radioactivity from the RUPTURED steam generator directly to the environment. This could be expected to occur when the main condenser is unavailable to accept the contaminated steam (i.e., SGTR with concurrent loss of offsite power and the RUPTURED steam generator is required for plant cooldown or a stuck open relief valve). If the main condenser is available, there may be releases via air ejectors, gland seal exhausters, and other similar controlled, and often monitored, pathways. These pathways do not meet the intent of a nonisolable release path to the environment. These minor releases are assessed using Abnormal Rad Levels / Radiological Effluent ICs. The two "loss" EALs described above could be considered redundant. This was recognized during the development process. The inclusion of an EAL that uses Emergency Procedure commonly used terms like "ruptured and faulted" adds to the ease of the classification process and has been included based on this human factor concern.

A pressure boundary leakage of 10 gpm is used as the threshold in IC SU5.1, PCS Leakage, and is deemed appropriate for this EAL. For smaller breaks, not exceeding the normal charging capacity threshold in PCS Barrier "Potential Loss" EAL #2 (PCS Leak Rate) or not resulting in ECCS actuation in EAL #3 (SG Tube Rupture), this EAL results in a UE. For larger breaks, PCS barrier EALs #2 and #3 would result in an Alert. For SG tube ruptures which may involve multiple steam generators or unisolable secondary line breaks, this EAL would exist in conjunction with

PCS barrier "Loss" EAL #3 and would result in a Site Area Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.

5. Containment Isolation Valve Status After Containment Isolation

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

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

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

6. Significant Radioactive Inventory in Containment

The 20,000 R/hr reading is a value which indicates significant fuel damage well in excess of the EALs associated with both loss of Fuel Clad and loss of PCS Barriers. [Ref. 4] A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant.

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

Containment radiation is indicated on Containment High Range Radiation Monitors, RIA-2321 and RIA-2322. The Containment High Range Radiation Monitors high alarm at 400 R/hr. [Ref. 5]

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

7. Other Indications

There are no "Loss" or potential loss EALs associated with this item.

8. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is lost or potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences. [As discussed in NEI 99-01 rev. 4]

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

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

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to EOP-9.0, "Functional Recovery." The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations. [Ref. 7, 8]

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

PNP Basis Reference(s):

1. PNP SAMG, "Phase 1.0 Initial Diagnosis"
2. EOP Setpoint Basis
3. GAMG, "Combustion Engineering Generic Accident Management Guidelines
4. NMC calculation EA-JLV-04-01, "Determination of Containment Radiation Monitor and Radiological Effluent Monitor EALs in Accordance with NEI 99-01 Revision 4"
5. ARP–33, "Auxiliary Systems Scheme EK-02 (C-11A)" Annunciator Nos. 1 and 2
6. EI-7.0, "Emergency Post Accident Sampling and Determination of Fuel Failure Using Dose Rates" Attachment 2, NSSS Sample Panel Piping Dose Rates
7. EOP-9.0, "Functional Recovery Procedure"
8. EOP-9.0, "Functional Recovery Procedure Basis"
9. ONP-23.1, "Primary Coolant Leak"
10. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident" dated November 1978
11. EOP-1.0, "Standard Post-Trip Actions"
12. FSAR Section 7.6, "CET System"
13. FSAR Section 5.8.1, "Containment Structure"
14. FSAR Section 9.10, "Chemical and Volume Control System"
15. EOP-4.0, "Loss of Coolant Accident Recovery Basis" Section 1.0 Introduction
16. FSAR Figure 14.18.1-1, "LOCA Containment Profile"
17. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13" Annunciator Nos. 40, 41, 42, 61
18. Vendor Drawing M1-BM Sh. 28, "RLI Display Panel"
19. FSAR Section 7.4.6.3 "Reactor Vessel Level Monitoring System RVLMS"
20. Vendor Drawing M-398 Sh. 1005 "Level Setting Diagram Reactor Vessel Level Monitoring System"
21. FSAR Section 7.4, "Instrumentation and Controls"

TABLE H-0

Recognition Category H

Hazards and Other Conditions Affecting Plant Safety

INITIATING CONDITION MATRIX

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

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HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU1

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Natural and Destructive Phenomena Affecting the PROTECTED AREA.

Operating Mode Applicability: All

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

HU1.1. Earthquake felt in plant as indicated by:

Report by Plant Personnel to on duty Control Room Personnel
AND
Confirmation from the National Earthquake Information Center

HU1.2. Report by plant personnel of tornado or high winds GREATER THAN 95 mph striking within PROTECTED AREA boundary.

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

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

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

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

- Emergency Diesel Generator Rooms
- Engineered Safeguards Rooms
- Auxiliary Feedwater Pump Room
- Switchgear Room 1C
- Screen House
- Component Cooling Water Pump Room

HU1.7. High lake level: ultimate heat sink level GREATER THAN 590 ft elevation affecting the PROTECTED AREA.

OR

Low lake level: ultimate heat sink level LESS THAN OR EQUAL TO 572 ft elevation.

Basis:

UE in this IC are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators. Areas identified in the EALs define the location of the event based on the potential for damage to equipment contained therein. Escalation of the event to an Alert occurs when the magnitude of the event is sufficient to result in damage to equipment contained in the specified location.

HU1.1. Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate. Method of detection is based on validation by a reliable source and operator assessment. As defined in the EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, a *"felt earthquake"* is:

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

Confirmation on seismic disturbances can be obtained from the National Earthquake Information Center, Denver, Colorado, 1-303-273-8500 (normal hours), or 1-303-273-8428 (off normal hours). Seismic instrumentation is available onsite for post emergency assessment of earthquakes. [Ref. 1, 2]

HU1.2 is based on the assumption that a tornado striking (touching down) or high winds within the PROTECTED AREA [Ref 4] may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. The high wind value is based on site-specific FSAR design basis of 100 mph [Ref. 3]. However, full-scale on the instrumentation used for classification is 100 mph. A value of 95 mph was chosen as the classification threshold, as this will still be on-scale. If such damage is confirmed visually or by other in-plant indications, the event may be escalated to Alert.

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

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

HU1.5 is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIREs and flammable gas build up are appropriately classified via HU2 and HU3. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant. This EAL is consistent with the definition of a UE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of the emergency classification is based on potential damage done by missiles generated by the failure or in conjunction with a steam generator tube rupture. These latter events would be classified by the radiological ICs or Fission Product Barrier ICs. [Ref. 5]

HU1.6 addresses the effect of flooding caused by internal events such as component failures, equipment misalignment, or outage activity mishaps. The site-specific areas include those areas that contain systems required for safe shutdown of the plant, that are not designed to be wetted or submerged. Escalation of the emergency classification is based on the damage caused or by access restrictions that prevent necessary plant operations or systems monitoring. [Ref. 6, 7, 8]

HU1.7 covers external flooding and low lake level. The nominal lake level is approximately 580 ft elevation.

The high lake water level threshold is ground level, which is approximately four feet below the design flood-seiche event level of 594.1 ft elevation. This level is obtained by adding a 10.9 ft onshore surge height to a maximum monthly mean lake level of 583.2 ft elevation. When lake water level approaches 590 ft elevation., ONP-12, "Acts of Nature", requires initiation of a plant shutdown [Ref. 2, 7, 9].

The low lake level is 216 in. below the Screen House floor level and ensures water level is well above the sluice gate opening which is at 568.25 ft elevation. Water level at this elevation ensures adequate NPSH to SWS pumps. The NPSH calculation assumes a minimum water level of 4 feet above the bottom of the pump suction bell which corresponds to an elevation of 557.25 ft. Violation of the SWS pump submergence requirement should never become a factor unless the Lake Michigan water level falls below the top of the sluice gate opening. ONP-6.1, "Loss of Service Water", requires tripping the reactor when service water bay level lowers below 572 ft. [Ref. 10]

PNP Basis Reference(s):

1. FSAR Section 2.4, "Seismicity"
2. ONP-12, "Acts of Nature"
3. FSAR Section 5.3.1.1, "Wind and Tornado Loadings – Design Parameters"
4. FSAR Figure 1-1, "Plant Area Plan"
5. FSAR Section 5.5, "Missile Protection"
6. DBD 7.08, "Plant Protection against Flooding"
7. FSAR Section 5.4, "Water Level Design"
8. FSAR Section 5.6, "Dynamic Effects Of Pipe Rupture"
9. FSAR Section 2.2.2, "General Lake Hydrology"
10. ONP-6.1, "Loss of Service Water"

HU2

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

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

Operating Mode Applicability: All

Emergency Action Level:

HU2.1. FIRE in buildings or areas contiguous to any of the following (Table H-1) areas not extinguished within 15 minutes of Control Room notification or verification of a Control Room alarm.

Table H-1 Plant VITAL AREAS
• Containment Structure • Auxiliary Building • Turbine Building • Screenhouse

Basis:

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

The intent of this 15 minute duration is to size the FIRE and to discriminate against small FIREs that are readily extinguished (e.g., smoldering waste paper basket). The Table H-1 areas are limited and apply to buildings and areas contiguous (in actual contact with or immediately adjacent) to plant VITAL AREAs or other significant buildings or areas. The intent of this IC is not to include buildings (i.e., warehouses) or areas that are not contiguous (in actual contact with or immediately adjacent) to plant VITAL AREAs. This EAL excludes FIREs within non-contiguous administration buildings, waste-basket FIREs and other small FIREs of no safety consequence.

Escalation to a higher emergency class is by IC HA2, "FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown." [Ref. 1]

PNP Basis Reference(s):

1. ONP-25.1, "Fire Which Threatens Safety-Related Equipment"

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU3

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

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

Operating Mode Applicability: All

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

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

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

Basis:

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

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

PNP Basis Reference(s):
None

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HU4

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

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

Operating Mode Applicability: All

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

HU4.1. Security Shift Leader reports ANY of the following:

- Suspected SABOTAGE device discovered within the plant PROTECTED AREA
- Suspected SABOTAGE device discovered outside the PROTECTED AREA or in the plant switchyard
- Confirmed tampering with safety-related equipment
- A HOSTAGE/EXTORTION situation that disrupts NORMAL PLANT OPERATIONS
- CIVIL DISTURBANCE or strike which disrupts NORMAL PLANT OPERATIONS
- Internal disturbance that is not short lived or that is not a harmless outburst involving ANY individuals within the PROTECTED AREA
- Malevolent use of a vehicle outside the PROTECTED AREA which disrupts NORMAL PLANT OPERATIONS

HU4.2. A credible site specific security threat notification.

Basis:

Reference is made to the Security Shift Leaders because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan. [Ref. 1]

HU4.1 is based on the PNP Site Security Plans. 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.

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

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

The determination of "credible" is made through use of information found in the Security Plan.

A higher initial classification could be made based upon the nature and timing of the threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification in accordance with the Security Plan and Emergency Plans.

PNP Basis Reference(s):

1. Security Plan (Safeguards Information)

HU5

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

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

Operating Mode Applicability: All

Emergency Action Level:

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

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the UE emergency class.

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

PNP Basis Reference(s):
None

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA1

Initiating Condition -- ALERT

Natural and Destructive Phenomena Affecting the Plant VITAL AREA.

Operating Mode Applicability: All

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

HA1.1. Seismic event GREATER THAN 0.1 g Operating Basis Earthquake (OBE) as indicated by VISIBLE DAMAGE to any of the following (Table H-2) plant structures / equipment.

Table H-2 VITAL AREAS/Safe Shutdown Equipment
• Containment Structure • Auxiliary Building • Turbine Building • Screenhouse • Safety Injection Refueling Water Tank (T-58) • Condensate Storage Tank (T-2) • Primary System Makeup Tank (T-81) • Fuel Oil Storage Tank (T-10A)

HA1.2. Tornado or high winds GREATER THAN 95 mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following (Table H-2) plant structures / equipment or Control Room indication of degraded performance of those systems.

HA1.3. Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following (Table H-2) plant structures or equipment therein or Control Room indication of degraded performance of those systems.

HA1.4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any of the following (Table H-2) plant areas.

HA1.5. Uncontrolled flooding in areas of the plant listed below that results in degraded safety system performance as indicated in the Control Room or that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment:

- Emergency Diesel Generator Rooms
- Engineered Safeguards Rooms
- Auxiliary Feedwater Pump Room

- Switchgear Room 1C
- Screen House
- Component Cooling Water Pump Room

HA1.6. The following occurrences within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to plant structures containing equipment necessary for safe shutdown, or has caused damage as evidenced by Control Room indication of degraded performance of those systems.

High lake level: ultimate heat sink level GREATER THAN 594 ft elevation

OR

Low lake level: ultimate heat sink level LESS THAN OR EQUAL TO 569 ft elevation.

Basis:

The Table H-2 areas contain systems and components required for the safe shutdown functions of the plant. The PNP safe shutdown analyses were consulted for equipment and plant areas required for the applicable mode. [Ref. 11] The EALs in this IC escalate from the UE EALs in HU1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by Control Room indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occur on the basis of other ICs (e.g., System Malfunction).

HA1.1 is based on the FSAR operating basis earthquake (OBE) of 0.1 g acceleration. Seismic events of this magnitude can result in a plant VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems. Confirmation on seismic disturbances can be obtained from the National Earthquake Information Center, Denver, Colorado, 1-303-273-8500 (normal hours), or 1-303-273-8428 (off normal hours). Seismic instrumentation is available onsite for post emergency assessment of earthquakes. [Ref.1, 2]

HA1.2 is based on the FSAR design basis sustained wind speed of 100 mph. However, full-scale on the instrumentation used for classification is 100 mph. A value of 95 mph was chosen as the classification threshold, as this will still be on-scale. [Ref. 3] Wind loads of this magnitude can cause damage to safety functions.

EAL #s HA1.2, HA1.3, HA1.4 and HA1.5 specify site-specific structures or areas containing systems and functions required for safe shutdown of the plant.

HA1.3 is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant. [Ref. 4]

HA1.4 is intended to address the threat to safety related equipment imposed by missiles generated by main turbine rotating component failures. Table H-2 lists areas that contain systems and components required for the safe shutdown functions of the plant.. This EAL is, therefore, consistent with the definition of an ALERT in that if missiles have damaged or penetrated areas

containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant. [Ref. 4, 5]

HA1.5 addresses the effect of internal flooding that has resulted in degraded performance of systems affected by the flooding, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant. This flooding may have been caused by internal events such as component failures, equipment misalignment, or outage activity mishaps. The site-specific areas includes those areas that contain systems required for safe shutdown of the plant, that are not designed to be wetted or submerged. [Ref 6, 7, 8]

HA1.6 covers flooding, low lake level or seiche. This EAL can be a precursor of more serious events. The high lake water level threshold is the design flood-seiche event level of 594.1 ft elevation. (rounded down to 594 ft). This level is obtained by adding a 10.9 ft onshore surge height to a maximum monthly mean lake level of 583.2 ft elevation. When lake water level approaches 590 ft elevation, ONP-12, "Acts of Nature", requires initiation of a plant shutdown. When lake water level exceeds 590 ft water may start to enter the Turbine Building/Screen House, ONP-12 requires tripping the reactor. [Ref. 2, 7, 9]

The low lake water level corresponds to the Screen House sluice gate opening of 568.25 ft elevation. (rounded to 569 ft). Water level at this elevation ensures adequate NPSH to SWS pumps. The NPSH calculation assumes a minimum water level of 4 feet above the bottom of the pump suction bell which corresponds to an elevation of 557.25 ft. Violation of the SWS pump submergence requirement should never become a factor unless the Lake Michigan water level falls below the top of the sluice gate opening. [Ref. 10].

PNP Basis Reference(s):

1. FSAR Section 2.4, "Seismicity"
2. ONP-12, "Acts of Nature"
3. FSAR Section 5.3.1.1, "Wind and Tornado Loadings – Design Parameters"
4. FSAR Figure 1-1, "Plant Area Plan"
5. FSAR Section 5.5, "Missile Protection"
6. DBD 7.08, "Plant Protection against Flooding"
7. FSAR Section 5.4, "Water Level Design"
8. FSAR Section 5.6, "Dynamic Effects Of Pipe Rupture"
9. FSAR Section 2.2.2, "General Lake Hydrology"
10. ONP-6.1, "Loss of Service Water"
11. ONP-25.1, "Fire Which Threatens Safety-Related Equipment"

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA2

Initiating Condition -- ALERT

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

Operating Mode Applicability: All

Emergency Action Level:

HA2.1. FIRE or EXPLOSION in any of the following (Table H-2) areas.

Table H-2 VITAL AREAS/Safe Shutdown Equipment
• Containment Structure • Auxiliary Building • Turbine Building • Screenhouse • Safety Injection Refueling Water Tank (T-58) • Condensate Storage Tank (T-2) • Primary System Makeup Tank (T-81) • Fuel Oil Storage Tank (T-10A)

AND

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

Basis:

The Table H-2 areas contain systems and components required for the safe shutdown functions of the plant. The PNP safe shutdown analyses were consulted for equipment and plant areas required for the applicable mode. This will make it easier to determine if the FIRE or EXPLOSION is potentially affecting one or more redundant trains of safety systems [Ref. 1]. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels / Radiological Effluent, or Emergency Director Judgment ICs.

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

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

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

The inclusion of a "report of VISIBLE DAMAGE" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the EXPLOSION with reports of evidence of damage is sufficient for declaration. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency Director with the resources needed to perform these damage assessments. The Emergency Director also needs to consider any security aspects of the EXPLOSIONs, if applicable. [Ref. 1]

PNP Basis Reference(s):

1. ONP-25.1, "Fire Which Threatens Safety-Related Equipment"

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA3

Initiating Condition -- ALERT

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

Operating Mode Applicability: All

Emergency Action Levels: (HA3.1 or HA3.2)

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

Table H-1 Plant VITAL AREAS
• Containment Structure • Auxiliary Building • Turbine Building • Screenhouse

HA3.2. Report or detection of gases in concentration greater than the LOWER FLAMMABILITY LIMIT within or contiguous to a VITAL AREA (Table H-1).

Basis:

This IC is based on gases that affect the safe operation of the plant. This IC applies to buildings and areas contiguous to plant VITAL AREAs or other significant buildings or areas [Ref. 1]. The intent of this IC is not to include buildings (e.g., warehouses) or other areas that are not contiguous or immediately adjacent to plant VITAL AREAs. It is appropriate that increased monitoring be done to ascertain whether consequential damage has occurred. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels / Radioactive Effluent, or Emergency Director Judgment ICs.

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

HA3.2 is met when the flammable gas concentration in a VITAL AREA or any building or area contiguous to a VITAL AREA exceed the LOWER FLAMMABILITY LIMIT. Flammable gasses, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to

repair equipment/components (acetylene - used in welding). This EAL addresses concentrations at which gases can ignite/support combustion. An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Once it has been determined that an uncontrolled release is occurring, then sampling must be done to determine if the concentration of the released gas is within this range.

PNP Basis Reference(s):

1. ONP-25.1, "Fire Which Threatens Safety-Related Equipment"

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA4

Initiating Condition -- ALERT

Confirmed Security Event in a Plant PROTECTED AREA.

Operating Mode Applicability: All

Emergency Action Levels: (HA4.1 or HA4.2)

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

HA4.2. Security Shift Leader reports any of the following:
- SABOTAGE device discovered in the plant PROTECTED AREA
- Standoff attack on the PROTECTED AREA by a HOSTILE FORCE (i.e., Sniper)
- ANY Security event of increasing severity that persists for GREATER THAN 30 min.:
  - Credible BOMB threats
  - HOSTAGE/EXTORTION
  - Suspicious FIRE OR EXPLOSION
  - Significant Security System Hardware Failure
  - Loss of Guard Post Contact

Basis:

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

The Security Plan identifies numerous events/conditions that constitute a threat/compromise to station security. Only those events that involve actual or potential substantial degradation to the level of safety of the plant need to be considered.

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

Reference is made to Security Shift Leaders because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan. [Ref. 1]

PNP Basis Reference(s):
1. Security Plan (Safeguards)

HA5

Initiating Condition -- ALERT

Control Room Evacuation Has Been Initiated.

Operating Mode Applicability: All

Emergency Action Level:

HA5.1. Entry into ONP-25.2, "Alternate Safe Shutdown" for Control Room evacuation.

Basis:

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facility is necessary. ONP-25.2, "Alternate Safe Shutdown", provides specific instructions for evacuating the Control Room and establishing plant control at remote Panels C-150 Auxiliary Hot Shutdown Panel/C-150A Auxiliary Hot Shutdown Monitoring Panel. Inability to establish plant control from outside the control room will escalate this event to a Site Area Emergency. [Ref. 1]

PNP Basis Reference(s):

1. ONP-25.2, "Alternate Safe Shutdown"

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HA6

Initiating Condition -- ALERT

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

Operating Mode Applicability: All

Emergency Action Level:

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

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency class.

PNP Basis Reference(s):
None

HS1

Initiating Condition – SITE AREA EMERGENCY

Confirmed Security Event in a Plant VITAL AREA.

Operating Mode Applicability: All

Emergency Action Levels: (HS1.1 or HS1.2)

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

HS1.2. Security Shift Leader reports any of the following:
- SABOTAGE discovered in a VITAL AREA.
- HOSTAGE/EXTORTION in a VITAL AREA.

Basis:

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

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

Reference is made to Security Shift Leaders because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security Plan. [Ref 1]

PNP Basis Reference(s):
1. Security Plan

HS2

Initiating Condition – SITE AREA EMERGENCY

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

Operating Mode Applicability: All

Emergency Action Level:

HS2.1. Control Room evacuation has been initiated.

AND

Control of the plant cannot be established per ONP- 25.2, "Alternate Safe Shutdown" procedure within 15 minutes.

Basis:

Expeditious transfer of safety systems has not occurred but fission product barrier damage may not yet be indicated. The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. The determination of whether or not control is established at the remote shutdown panel is based on Emergency Director (ED) judgment. The ED is expected to make a reasonable, informed judgment within the time for transfer that the operator has control of the plant from the remote shutdown panel.

ONP-25.2, Alternate Safe Shutdown, provides specific instructions for evacuating the Control Room and establishing plant control at the remote Panels C-150 Auxiliary Hot Shutdown Panel / C-150A Auxiliary Hot Shutdown Monitoring Panel. ONP-25.2 also specifies required times for completing individual operator actions to control the plant. The required times for completing all of these actions are greater than 15 minutes with the exception of isolating atmospheric steam dumps. Steam dumps need to be isolated within six minutes to avoid overcooling the PCS. However, this action would be completed prior to establishing control at C-150/C-150A. The 15 minute limit is established to ensure control is established at C-150/C-150A in sufficient time to allow completion of the remaining actions in ONP-25.2. [Ref. 1]

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

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

PNP Basis Reference(s):

1. ONP-25.2, "Alternate Safe Shutdown"

HS3

Initiating Condition – SITE AREA EMERGENCY

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

Operating Mode Applicability: All

Emergency Action Level:

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

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency class description for Site Area Emergency.

PNP Basis Reference(s):
None

HG1

Initiating Condition – GENERAL EMERGENCY

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

Operating Mode Applicability: All

Emergency Action Level:

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

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

OR

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

Basis:

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

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

Loss of physical control of the control room or remote shutdown capability alone may not prevent the ability to maintain safety functions per se.[Ref. 1]

PNP Basis Reference(s):

1. Security Plan (Safeguards)

HG2

Initiating Condition – GENERAL EMERGENCY

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

Operating Mode Applicability: All

Emergency Action Level:

HG2.1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the General Emergency class.

PNP Basis Reference(s):
None

Table S-0

Recognition Category S

System Malfunction

INITIATING CONDITION MATRIX

UE	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
SU1 Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SA5** AC power capability to essential busses reduced to a single power source for GREATER THAN 15 minutes such that any additional single failure would result in station blackout. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SS1** Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SG1** Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*
	SA2 Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram Was Successful. *Op. Modes: Power Operation, Startup, Hot Standby*	**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. *Op. Modes: Power Operation, Startup*	**SG2** Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core. *Op. Modes: Power Operation, Startup*
SU2 Inability to Reach Required Shutdown Within Technical Specification Limits. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SA3** *Deleted*	**SS4** Complete Loss of Heat Removal Capability. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	
SU3 UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for GREATER THAN15 Minutes *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SA4** UNPLANNED Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a SIGNIFICANT TRANSIENT in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	**SS6** Inability to Monitor a SIGNIFICANT TRANSIENT in Progress. *Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*	

Recognition Category S

System Malfunction

INITIATING CONDITION MATRIX

SU7 *Deleted*

SU4 Fuel Clad Degradation.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SU5 PCS Leakage.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SU6 UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SU8 Inadvertent Criticality.
Op Modes: Hot Standby, Hot Shutdown

SA1 *Deleted*

SS3 Loss of All Vital DC Power.
Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown

SS5 *Deleted*

SYSTEM MALFUNCTION

SU1

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

Loss of All Offsite Power to Essential Busses forGREATER THAN 15 Minutes.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SU1.1. Loss of all offsite power to both Vital 2400 VAC busses 1C and 1D for GREATER THAN 15 minutes.

AND

Both emergency diesel generators are supplying power to both Vital 2400 VAC busses 1C and 1D.

Basis:

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

The 2400 VAC system consists of Safeguard Transformer 1-1, Station Power Transformer 1-2, Start Up Transformer 1-2, four 2400 VAC busses (1C, 1D, 1E and Safeguards). Busses 1C and 1D are the VITAL (essential) busses that supply power to engineered safeguards loads. [Ref. 1] Diesel Generator 1-1 will start when an undervoltage is sensed on 2400 Volt Bus 1C and Diesel Generator 1-2 will start when an undervoltage is sensed on 2400 Volt Bus 1D. [Ref. 2]

PNP Basis Reference(s):

1. FSAR Section 8.3.2, "Electrical Systems - 2,400 Volt System"
2. FSAR Section 8.4.1, "Electrical Systems - Emergency Generators"

SU2

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Inability to Reach Required Shutdown Within Technical Specification Limits.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SU2.1. Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time.

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required shutdown mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the PNP Technical Specifications requires a one hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate UE is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of a UE is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed. Other required Technical Specification shutdowns that involve precursors to more serious events are addressed by other System Malfunction, Hazards, or Fission Product Barrier Degradation ICs.

PNP Basis Reference(s):

1. None

SYSTEM MALFUNCTION

SU3

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for GREATER THAN 15 Minutes

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SU3.1. UNPLANNED loss of most or all annunciators or indicators associated with safety systems for GREATER THAN 15 minutes on the following:

- EC-02/12 – Reactor/PCPs
- EC-03/13 – Safety Injection/Containment Cooling & Isolation systems / Ventilation
- EC-04 – Electrical
- EC-06 – RPS
- EC-08 – Service Water/Component Cooling
- EC-11 (Rear) – Rad Monitors
- EC-11A (Front) – Control Room HVAC / Reactor Vessel Level / Core Exit Thermocouples
- EC-27 - Thermal Margin Monitors

Basis:

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer based indication equipment is considered (e.g., PPC). [Ref. 1]

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions.

It is further recognized that plant design provides redundant safety system indication powered from separate uninterruptable power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on SU2 "Inability to Reach Required Shutdown Within Technical Specification Limits."

The specified annunciators [Ref. 2, 3, 4, 5, 6, 7] or indicators [Ref. 8] for this EAL include those identified in the Off Normal Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. Due to the limited number of safety systems in operation during cold shutdown, refueling, and defueled modes, no IC is indicated during these modes of operation.

This UE will be escalated to an Alert if a transient is in progress during the loss of annunciation or indication.

PNP Basis Reference(s):

1. FSAR Section 7.6.2.5, "Instrumentation and Controls - Palisades Plant Computer"
2. ARP-4, "Primary System Volume Level Pressure Scheme EK-07 (C-12)"
3. ARP-5, "Primary Coolant Pump Steam Generator and Rod Drives Scheme EK-09 (C-12)"
4. ARP-7, "Auxiliary Systems Scheme EK-11 (C-13)"
5. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (C-13)"
6. ARP-21, "Reactor Protective System Scheme EK-06 (C-06)"
7. ARP-33. "Auxiliary Systems Scheme EK-02 (C-11A)"
8. P&ID Equipment Location Reactor and Aux. Bldg. M-4 (C-5/6) [For EC-02, 03, 04, 06, 08, 11, 11A, 12, 13 & 27]

SYSTEM MALFUNCTION

SU4

Initiating Condition – NOTIFICATION OF UNUSUAL EVENT

Fuel Clad Degradation.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Levels: (SU4.1 or SU4.2)

SU4.1. Any of the following radiation monitors with a VALID PPC urgent alarm indicating fuel clad degradation GREATER THAN Technical Specification allowable limits.

Containment Isolation High Radiation Monitor - RIA-1805
Containment Isolation High Radiation Monitor - RIA-1806
Containment Isolation High Radiation Monitor - RIA-1807
Containment Isolation High Radiation Monitor - RIA-1808

SU4.2. Coolant sample activity GREATER THAN OR EQUAL TO 40 µCi/gm dose equivalent I-131 indicating fuel clad degradation.

Basis:

This IC is included as a UE because it is considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. SU4.1 addresses site-specific radiation monitor readingsthat provide indication of fuel clad integrity. The Containment Isolation High Radiation Monitors will initiate containment isolation on 2 out of 4 coincidence logic at a high alarm reading of 10 R/hr [Ref. 1]. Assured effectiveness of these monitors has been verified by actual testing for the case of core damage assuming approximately 1% failed fuel without LOCA [Ref. 2]. A special test conducted in 1980 in response to NUREG-0737 verified that with approximately 1% failed fuel the Containment Isolation High Radiation monitors would indicate 4.5 Rem/hour [Ref. 3]. The PPC urgent alarm is set at one tenth of that value, 0.45 Rem/hour, indicating fuel clad degradation [Ref. 4].

SU4.2 addresses coolant samples exceeding coolant technical specifications for iodine spike. The Technical Specification safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the 10 CFR 100 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit of 1.0 µCi/gm is permissible, if the activity levels do not exceed the limit of 40 µCi/gm for more than 48 hours.

This is acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of a SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits. [Ref. 5, 6]

Escalation of this IC to the Alert level is via the Fission Product Barrier Degradation Monitoring ICs. Though the referenced Technical Specification limits are mode dependent, it is appropriate that the

EAL's be applicable in all modes, as they indicate a potential degradation in the level of safety of the plant. The companion IC to SU4 for the Cold Shutdown/Refueling modes is CU5.

PNP Basis Reference(s):

1. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (EC-13)" (Window #63)
2. FSAR Chapter 7, "Instrumentation and Controls", Section 7.3, Engineered Safeguards Controls, Step 7.3.3.3, "Design Analysis"
3. Consumers Power Company, Response to NUREG-0737, December 19, 1980 (Item II.E.4.3 - Special Test of April 15, 1980)
4. PPC Containment Isolation High Radiation Monitor Urgent Alarm Setpoints
5. Technical Specifications 3.4.16, "PCS Specific Activity"
6. Technical Specifications B 3.4.16, "PCS Specific Activity"

SYSTEM MALFUNCTION

SU5

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

PCS Leakage.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Levels: (SU5.1 or SU5.2)

SU5.1. Unidentified or pressure boundary leakage GREATER THAN 10 gpm.

SU5.2. Identified leakage GREATER THAN 25 gpm.

Basis:

This IC is included as a UE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is observable with normal control room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. In either case, escalation of this IC to the Alert level is via Fission Product Barrier Degradation ICs.

PNP Basis Reference(s):

None

SYSTEM MALFUNCTION

SU6

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Levels: (SU6.1 or SU6.2)

SU6.1. Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations.

Table C-1 Onsite Communications Systems
• Telephone system
• Onsite/offsite radio system
• Public address system

SU6.2. Loss of all Table C-2 offsite communications capability.

Table C-2 Offsite Communications Systems
• Telephone system
• Power failure phones
• FTS phone system
• Satellite phone

Basis:

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

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

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

Table C-2 offsite communications loss [Ref. 1, 2, 3] encompasses the loss of all means of communications with offsite authorities. This includes the ENS, commercial telephone lines, telecopy transmissions, and dedicated phone systems.

PNP Basis Reference(s):

1. FSAR Section 7.7.8, "Instrumentation and Controls - In-Plant Communication System"
2. SOP-31, "Plant Lighting and Communications"
3. EI-15.2, "Communications Tests"

SYSTEM MALFUNCTION

SU8

Initiating Condition -- NOTIFICATION OF UNUSUAL EVENT

Inadvertent Criticality.

Operating Mode Applicability Hot Standby
Hot Shutdown

Emergency Action Level: (SU8.1)

SU8.1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

This IC addresses inadvertent criticality events. While the primary concern of this IC is criticality events that occur in Cold Shutdown or Refueling modes (NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States), the IC is applicable in other modes in which inadvertent criticalities are possible. This IC indicates a potential degradation of the level of safety of the plant, warranting an UE classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated). The Cold Shutdown/Refueling IC is CU8.

This condition can be identified using the startup rate monitor. The term "sustained" is used in order to allow exclusion of expected short term positive startup rates from planned control rod movements such as shutdown bank withdrawal. These short term positive startup rates are the result of the rise in neutron population due to subcritical multiplication.

Escalation would be by the Fission Product Barrier Matrix, as appropriate to the operating mode at the time of the event, or by Emergency Director Judgment.

PNP Basis Reference(s):

None

SYSTEM MALFUNCTION

SA2

Initiating Condition – ALERT

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 Successful.

Operating Mode Applicability: Power Operation
Startup
Hot Standby

Emergency Action Level:

SA2.1.

Indication(s) exist that a Reactor Protection System setpoint was exceeded

AND

RPS automatic trip did NOT occur

AND

A successful manual trip occurred from:

- EC-02 Reactor Trip pushbutton

  OR

- EC-06 Reactor Trip pushbutton

  OR

- Reactor tripped from an ATWS trip signal

Basis:

This condition indicates failure of the automatic protection system to trip the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or PCS. Reactor protection system setpoint being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because failure of the automatic protection system is the issue. A manual scram is any set of actions by the reactor operator(s) at the reactor control console which causes control rods to be rapidly inserted into the core and brings the reactor subcritical (e.g., reactor trip button, Alternate Rod Insertion). Failure of manual scram would escalate the event to a Site Area Emergency.

EOP-1.0, "Standard Post Trip Actions" instructs the operator to push both reactor trip pushbuttons on Control Room panels EC-02 and EC-06. Manual trips are not considered successful if action taken away from the Control Room panels EC-02 and EC-06 is required to trip the reactor. If

opening Control Rod Drive clutch power feeder breakers 42-1 RPS and 42-2 RPS with a standing reactor trip signal present, placing all CRD clutch power toggle switches to CLUTCH OFF, or emergency borating are required to shutdown the reactor, manual trips are not considered to be successful and classification under EAL SS2.1 or SG2.1 would be required [Ref. 1, 2]. If a manual reactor trip is required to achieve reactor shutdown, a condition that is more than a potential degradation of a safety system has occurred in that a front line automatic protection system (RPS) did not function in response to a plant transient. Plant safety has thus been compromised and design limits of the fuel may have been exceeded. An Alert is indicated because conditions exist that lead to potential loss of Fuel Cladding or Primary Coolant System barrier integrity. The RPS trip setpoint being exceeded, rather than a limiting safety system setpoint, is specified here because the 'first line of defense' automatic protection system has failed.

A reactor trip from the ATWS trip signal warrants an Alert classification, as it is a 'second line of defense' trip equivalent to the manual pushbutton trips. The ATWS trip circuitry uses a diverse trip methodology. This trip methodology accomplishes a reactor trip via the same circuitry as the EC-06 Reactor Trip manual pushbutton [Ref. 3]. An Alert classification is appropriate as the ATWS trip setpoint (PCS pressure greater than 2375 psia) is designed to avoid overlap with the RPS trips or PCS safety valves [Ref. 4].

PNP Basis Reference(s):

1. EOP-1.0, "Standard Post-Trip Actions"
2. EOP-1.0, "Standard Post-Trip Actions Basis"
3. FSAR Figure 7-1 - RPS Block Diagram
4. FSAR Chapter 7, Section 7.2, "Reactor Protection System"

SYSTEM MALFUNCTION

SA4

Initiating Condition – ALERT

UNPLANNED Loss of Most or All Safety System Annunciation or Indication inControl Room With Either (1) a SIGNIFICANT TRANSIENT in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SA4.1. UNPLANNED loss of most or all annunciators or indicators associated with safety systems for GREATER THAN 15 minutes on the following:

- EC-02/12 – Reactor/PCPs
- EC-03/13 – Safety Injection/Containment Cooling & Isolation systems / Ventilation
- EC-04 – Electrical
- EC-06 – RPS
- EC-08 – Service Water/Component Cooling
- EC-11 (Rear) – Rad Monitors
- EC-11A (Front) – Control Room HVAC / Reactor Vessel Level / Core Exit Thermocouples
- EC-27 - Thermal Margin Monitors

AND

Either of the following:

a. A SIGNIFICANT TRANSIENT is in progress.

OR

b. Compensatory non-alarming indications are unavailable.

Basis:

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a transient. Recognition of the availability of computer based indication equipment is considered (e.g., PPC) [Ref. 1].

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

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that plant design provides redundant safety system indication powered from separate uninterruptable power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on SU2 "Inability to Reach Required Shutdown Within Technical Specification Limits."

The specified annunciators [Ref. 2, 3, 4, 5, 6, 7] or indicators [Ref. 8] for this EAL include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

"Compensatory non-alarming indications" include the PPC, plant recorders, or plant instrument displays in the control room. If both a major portion of the annunciation system and all computer monitoring are unavailable, the Alert is required. Due to the limited number of safety systems in operation during cold shutdown, refueling and defueled modes, no IC is indicated during these modes of operation.

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress.

PNP Basis Reference(s):

1. FSAR Section 7.6.2.5, "Instrumentation and Controls - Palisades Plant Computer"
2. ARP-4, "Primary System Volume Level Pressure Scheme EK-07 (C-12)"
3. ARP-5, "Primary Coolant Pump Steam Generator and Rod Drives Scheme EK-09 (C-12)"
4. ARP-7, "Auxiliary Systems Scheme EK-11 (C-13)"
5. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (C-13)"
6. ARP-21, "Reactor Protective System Scheme EK-06 (C-06)"
7. ARP-33, "Auxiliary Systems Scheme EK-02 (C-11A)"
8. P&ID Equipment Location Reactor and Aux. Bldg. M-4 (C-5/6) [For EC-02, 03, 04, 06, 08, 11, 11A, 12, 13 & 27]

SYSTEM MALFUNCTION

SA5

Initiating Condition – ALERT

AC power capability to essential busses reduced to a single power source for GREATER THAN 15 minutes such that any additional single failure would result in station blackout.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SA5.1. AC power capability to Vital 2400 VAC ~~site specific essential~~ busses 1C and 1D reduced to only one of the following sources for GREATER THAN 15 minutes

- Safeguard Transformer 1-1
- Start Up Transformer 1-2
- Station Power Transformer 1-2
- 1-1 Emergency Diesel Generator
- 1-2 Emergency Diesel Generator

AND

Any additional single failure will result in station blackout.

Basis:

This IC and the associated EALs are intended to provide an escalation from IC SU1, "Loss of All Offsite Power To Essential Busses for Greater Than 15 Minutes." The condition indicated by this IC is the degradation of the offsite and onsite power systems such that any additional single failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of one emergency diesel generator to supply power to its emergency busses. Another related condition could be the loss of all offsite power and loss of onsite emergency diesels with only one train of emergency busses being backfed from the unit main generator, or the loss of onsite emergency diesels with only one train of emergency busses being backfed from offsite power.

The 2400 VAC system consists of Safeguard Transformer 1-1, Station Power Transformer 1-2, Start Up Transformer 1-2, four 2400 VAC busses (1C, 1D, 1E and Safeguards). Busses 1C and 1D are the VITAL (essential) busses that supply power to engineered safeguards loads. [Ref. 1] Diesel Generator 1-1 will start when an undervoltage is sensed on 2400 Volt Bus 1C and Diesel Generator 1-2 will start when an undervoltage is sensed on 2400 Volt Bus 1D. [Ref. 2]

The subsequent loss of the single remaining power source would escalate the event to a Site Area Emergency in accordance with IC SS1, "Loss of All Offsite and Loss of All Onsite AC Power to Essential Busses."

PNP Basis Reference(s):

1. FSAR Section 8.3.2, "Electrical Systems - 2,400 Volt System"
2. FSAR Section 8.4.1, "Electrical Systems - Emergency Generators"

SYSTEM MALFUNCTION

SS1

Initiating Condition -- SITE AREA EMERGENCY

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

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SS1.1. Loss of all offsite power to Vital 2400 VAC busses 1C and 1D

AND

Failure of both emergency diesel generators to supply power to Vital 2400 VAC busses .

AND

Failure to restore power to at least one Vital 2400 VAC bus within 15 minutes from the time of loss of both offsite and onsite AC power.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including Shutdown Cooling, Safety Injection, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will cause core uncovering and loss of containment integrity, thus this event can escalate to a General Emergency. The time duration was selected to exclude transient or momentary power losses.

The 2400 VAC system consists of Safeguard Transformer 1-1, Station Power Transformer 1-2, Start Up Transformer 1-2, four 2400 VAC busses (1C, 1D, 1E and Safeguards). Busses 1C and 1D are the VITAL (essential) busses that supply power to engineered safeguards loads. [Ref. 1] Diesel Generator 1-1 will start when an undervoltage is sensed on 2400 Volt Bus 1C and Diesel Generator 1-2 will start when an undervoltage is sensed on 2400 Volt Bus 1D. [Ref. 2] Escalation to General Emergency is via Fission Product Barrier Degradation or IC SG1, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power."

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of AC power to Vital 2400 VAC busses. Even though a Vital 2400 VAC bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus then the bus should not be considered operable. If this bus was the only energized bus then a Site Area Emergency per SS1 should be declared.

PNP Basis Reference(s):

1. FSAR Section 8.3.2, "Electrical Systems - 2,400 Volt System"
2. FSAR Section 8.4.1, "Electrical Systems - Emergency Generators"

SYSTEM MALFUNCTION

SS2

Initiating Condition -- SITE AREA EMERGENCY

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.

Operating Mode Applicability: Power Operation
Startup

Emergency Action Level:

SS2.1. Indication(s) exist that automatic and manual trip were NOT successful.

Basis:

Automatic and manual trip are not considered successful if action away from the reactor control console was required to trip the reactor.

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency is indicated because conditions exist that lead to imminent loss or potential loss of both fuel clad and PCS. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response. Escalation of this event to a General Emergency would be via Fission Product Barrier Degradation or Emergency Director Judgment ICs.

Manual trips are not considered successful if action taken away from the Control Room panels EC-02 and EC-06 is required to trip the reactor. If opening Control Rod Drive clutch power feeder breakers 42-1 RPS and 42-2 RPS with a standing reactor trip signal present, placing all CRD clutch power toggle switches to CLUTCH OFF, or emergency borating are required to shutdown the reactor, manual trips are not considered to be successful [Ref. 1, 2]. If these actions prove unsuccessful in shutting down the reactor, escalation via EAL SG2.1 would be required. Negative startup rate and lowering reactor power are indications of successful reactor shutdown and should be observed following any reactor trip from power.

PNP Basis Reference(s):

1. EOP-1.0, "Standard Post Trip Actions"
2. EOP-1.0, "Standard Post Trip Actions Basis"

SYSTEM MALFUNCTION

SS3

Initiating Condition -- SITE AREA EMERGENCY

Loss of All Vital DC Power.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SS3.1. Loss of all vital DC power based on LESS THAN 105 VDC on DC busses No. 1, ED-10 and No.2, ED-20 for GREATER THAN 15 minutes.

Basis:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. Escalation to a General Emergency would occur by Abnormal Rad Levels/Radiological Effluent, Fission Product Barrier Degradation, or Emergency Director Judgment ICs. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Station Battery #1 and Station Battery #2.have ample capacity to supply required DC loads (DC busses No. 1, ED-10 and No.2, ED-20) and preferred AC loads during a complete loss of AC power for at least four hours, assuming neither emergency diesel generator is available. The batteries are designed to furnish their maximum load down to an operating temperature of 70 degrees F without dropping below 105 VDC, and the equipment supplied by the batteries is capable of operating satisfactorily at this voltage rating. 105 VDC represents ~80% of the manufacturers' rating for battery capacity of a nominal 131 VDC. [Ref. 1]

PNP Basis Reference(s):

1. FSAR Section 8.4.2, "Electrical Systems - Station Batteries"

SYSTEM MALFUNCTION

SS4

Initiating Condition – SITE AREA EMERGENCY

Complete Loss of Heat Removal Capability.

Operating Mode Applicability:

Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SS4.1. Loss of core cooling and heat sink.

SS4.

Basis:

This EAL addresses complete loss of functions, including ultimate heat sink, required for hot shutdown with the reactor at pressure and temperature. Reactivity control is addressed in other EALs.

Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted. Escalation to General Emergency would be via Abnormal Rad Levels / Radiological Effluent, Emergency Director Judgment, or Fission Product Barrier Degradation ICs.

PNP Basis Reference(s):

None

Initiating Condition – SITE AREA EMERGENCY

Inability to Monitor a SIGNIFICANT TRANSIENT in Progress.

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SS6.1. Loss of most or all annunciators associated with safety systems on the following:

- EC-02/12 – Reactor/PCPs
- EC-03/13 – Safety Injection/Containment Cooling & Isolation systems / Ventilation
- EC-04 – Electrical
- EC-06 – RPS
- EC-08 – Service Water/Component Cooling
- EC-11 (Rear) – Rad Monitors
- EC-11A (Front) – Control Room HVAC / Reactor Vessel Level / Core Exit Thermocouples
- EC-27 - Thermal Margin Monitors

AND

SIGNIFICANT TRANSIENT in progress.

AND

Compensatory non-alarming indications are unavailable.

AND

Indications needed to monitor the ability to shut down the reactor, maintain the core cooled, maintain the reactor coolant system intact, and maintain containment intact are unavailable.

Basis:

This IC and its associated EAL are intended to recognize the inability of the control room staff to monitor the plant response to a transient. A Site Area Emergency is considered to exist if the control room staff cannot monitor safety functions needed for protection of the public.

SIGNIFICANT TRANSIENT is an UNPLANNED event involving one ore more of the following: (1) turbine runback GREATER THAN 25% thermal reactor power, (2) electrical load rejection GREATER

THAN 25% full electrical load, (3) Reactor Trip, (4) Safety Injection Activation, or (5) thermal power oscillations GREATER THAN 10%.

"Compensatory non-alarming indications" include the PPC [Ref. 1], plant recorders, or plant instrument displays in the control room. [Ref. 8]

Indications needed to monitor safety functions necessary for protection of the public include control room indications, computer generated indications and dedicated annunciation capability [Ref. 2, 3, 4, 5, 6, 7]. The specific indications are those used to monitor the ability to shut down the reactor, maintain the core cooled, to maintain the reactor coolant system intact, and to maintain containment intact.

"Planned" and "UNPLANNED" actions are not differentiated since the loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

PNP Basis Reference(s):

1. FSAR Section 7.6.2.5, "Instrumentation and Controls - Palisades Plant Computer"
2. ARP-4, "Primary System Volume Level Pressure Scheme EK-07 (C-12)"
3. ARP-5, "Primary Coolant Pump Steam Generator and Rod Drives Scheme EK-09 (C-12)"
4. ARP-7, "Auxiliary Systems Scheme EK-11 (C-13)"
5. ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13 (C-13)
6. ARP-21, "Reactor Protective System Scheme EK-06 (C-06)"
7. ARP-33, "Auxiliary Systems Scheme EK-02 (C-11A)"
8. P&ID Equipment Location Reactor and Aux. Bldg. M-4 (C-5/6) [For EC-02, 03, 04, 06, 08, 11, 11A, 12, 13 & 27]

SYSTEM MALFUNCTION

SG1

Initiating Condition -- GENERAL EMERGENCY

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

Operating Mode Applicability: Power Operation
Startup
Hot Standby
Hot Shutdown

Emergency Action Level:

SG1.1. Loss of all offsite power to Vital 2400 VAC busses 1C and 1D

AND

Failure of both emergency diesel generators to supply power to Vital 2400 VAC busses.

AND

Either of the following:

a. Restoration of at least one Vital 2400 VAC bus within 4 hours is NOT likely

OR

b. Continuing degradation of core cooling based on Fission Product Barrier monitoring as indicated by Average of qualified CETs GREATER THAN 700 degrees F.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including Shutdown Cooling, Safety Injection, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will lead to loss of fuel clad, PCS, and containment. The 4 hours to restore AC power is based on the site blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout," [Ref 1]. Appropriate allowance for offsite emergency response including evacuation of surrounding areas should be considered. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.

This IC is specified to assure that in the unlikely event of a prolonged station blackout, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory.

The 2400 VAC system consists of Safeguard Transformer 1-1, Station Power Transformer 1-2, Start Up Transformer 1-2, four 2400 VAC busses (1C, 1D, 1E and Safeguards). Busses 1C and 1D are the VITAL (essential) busses that supply power to engineered safeguards loads. [Ref. 2]

Diesel Generator 1-1 will start when an undervoltage is sensed on 2400 Volt Bus 1C and Diesel Generator 1-2 will start when an undervoltage is sensed on 2400 Volt Bus 1D. [Ref. 3]

The likelihood of restoring at least one emergency bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.
In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly (s)he may need to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is imminent? (Refer to Table F-1 for more information.) This threshold (CETs in excess of 700 degrees F) indicates loss of inventory control resulting in significant core exit superheating. The highest CET temperature expected for any FSAR analysis accident is 700 degrees F and is a limiting condition to remain in an Optimal Recovery Procedure [Ref. 4].

2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.

PNP Basis Reference(s):

1. FSAR Section 8.4.2, "Electrical Systems - Station Batteries"
2. FSAR Section 8.3.2, "Electrical Systems - 2,400 Volt System"
3. FSAR Section 8.4.1, "Electrical Systems - Emergency Generators"
4. EOP Setpoint Basis

SYSTEM MALFUNCTION

SG2

Initiating Condition -- GENERAL EMERGENCY

Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core.

Operating Mode Applicability: Power Operation
Startup

Emergency Action Level:

SG2.1. Indications exist that automatic and manual trip were not successful.

AND

Either of the following:

a. Average of qualified CETs GREATER THAN 1200 degrees F.

OR

b. Core and PCS Heat Removal safety function status acceptance criteria cannot be met.

Basis:

Automatic and manual trip are not considered successful if action away from the reactor control console is required to trip the reactor.

Under the conditions of this IC and its associated EALs, the efforts to bring the reactor subcritical have been unsuccessful and, as a result, the reactor is producing more heat than the maximum decay heat load for which the safety systems were designed. Although there are capabilities away from the reactor control console, such as emergency boration the continuing temperature rise indicates that these capabilities are not effective. This situation could be a precursor for a core melt sequence.

The combination of failure of both automatic and manual reactor 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 PCS barriers. Note that the plant operating mode changes to Hot Shutdown as soon as a successful reactor trip occurs. Since this EAL is applicable only to Power Operation and Startup modes, escalation to the General Emergency classification is not appropriate under this EAL after the reactor is successfully tripped. [Ref. 1, 2]

The extreme challenge to the ability to cool the core is intended to mean that the core exit temperatures are at or approaching 1200 degrees F. The Average of the qualified Core Exit Thermocouples (CETs) provide an indirect indication of fuel cladding temperature by measuring the temperature of the primary coolant that leaves the core region. The extreme challenge to core cooling threshold temperature of 1200 degrees F is consistent with CEOG Generic Accident Management Guidelines for Phase 1, "Initial Diagnosis" for core exit temperature. Although clad

rupture due to high temperature is not expected for CET readings less than the threshold, temperatures of this magnitude signal severe superheating of the primary coolant and core uncovery. [Ref. 3]

Another consideration is the inability to initially remove heat during the early stages of this sequence. If emergency feedwater flow is insufficient to remove the amount of heat required by design from at least one steam generator, an extreme challenge should be considered to exist. Core and PCS Heat Removal safety function status acceptance criteria are specified in the EOPs. [Ref. 4]In the event either of these challenges exist at a time that the reactor has not been brought below the power associated with the safety system design a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier matrix declaration to permit maximum offsite intervention time.

PNP Basis Reference(s):

1. EOP-1.0, "Standard Post Trip Actions"
2. EOP-1.0, "Standard Post Trip Actions Basis"
3. CEOG Generic Accident Management Guidelines - Phase 1, "Initial Diagnosis"
4. EOP-9.0 "Functional Recovery Procedure" Basis (HR-1, HR-2 & HR-3)

ATTACHMENT 3

JUSTIFICATION MATRIX

174 Pages Follow

Table of Contents

Summary

Justification Matrix Format/Content

The justification matrices contain justifications supporting the implementation of the PNP plant specific EALs and the EAL Technical Basis Document, relating differences from the NEI 99-01 revision 4, "Methodology For Development of Emergency Action Levels." The justification matrices address the inputted site-specific information, any differences and/or deviations identified in the initiating conditions, EAL thresholds, and EAL basis.

The matrices are divided into the following sections and two addendums:

- Generic Differences Justifications
- Abnormal Rad Levels / Radiological Effluent Justifications
- Cold Shutdown / Refueling System Malfunctions Justifications
- ISFSI Justifications
- Fission Product Barrier Justifications
- Hazards Justifications
- System Malfunctions

Addendum Number 1, "Justification for RVLMS Levels"
Addendum Number 2, "EAL RA1.2 Deviation Justification"

The generic difference justifications provide the justifications related to the PNP numbering scheme changes from the NEI guideline and other changes that were globally changed throughout many of the EALs. Uses of the generic justifications were noted within the justification tables for the specific ICs and EALs where they were used. The TBD basis discussion used many of these generic differences, although they were not noted in basis justification.

The justification matrix format is as follows:

- Initiating Condition: comparison of NEI IC to PNP IC, description of site-specific information, discussion of differences and deviations.
- Associated EALs: NEI to PNP EAL comparison, description of site-specific information, discussion of differences and deviations.
- Basis Justification: PNP specific additions/deletions and justification for each, discussion of differences and deviations.

This format is consistent throughout the matrices with the exception of the FPB category. For the FPB category the justification for the basis was placed in a separate table for each FPB at the end of the EAL threshold justifications.

Overview of Deviations

Palisades has three plant design issues affecting 11 EALs. Two of the design issues are described in the two addendums to the justification matrices. These two addendums provide a detailed explanation of the site-specific design issues as they relate to the implementation of the NEI guidance.

The specific EALs with deviations related to the three plant design issues are as follows:

1. PNP RVLMS Design Addendum No. 1, " insert title "
   PNP RVLMS design is unable to detect levels that specifically match the NEI specified levels. Based on the indications available, PNP has applied the NEI methodology to allow indications for each level of classification. However, the levels that can be detected result in conservative classification calls. The detailed discussion is contained in Addendum 1, "Justification for RVLMS Levels." Affected EALs: CA1.1, CA2.1, CS1.1, CS1.2, CS2.1, CS2.2, CG1.2, FPB FC PL#4, and FPB CMT PL#3.

2. Offsite Release Escalation Scheme (Addendum No. 2):
   Affected EAL: RA1.2
   Discussion: The deviation only applies to the two Radioactive Gaseous Effluent Monitors (RGEM) listed in the EAL (RIA-2326 and RIA-2327). The NEI guidance for RA1.2 is a release that exceeds 200 times the ODCM. Application of 200 times the ODCM Limit for the RGEM monitors results in values that exceed the offsite dose limits specified for RS1.1. Therefore if the 200 times ODCM limit were used, an Alert would never be declared since the SAE threshold would already be exceeded. To maintain intervals between classifications and declare the Alert at an appropriate level, PNP selected limits for these monitors, which are less than 200 times the ODCM Limits. The detailed discussion is contained in Addendum 2, "EAL RA1.2 Deviation Justification."

3. Failed Fuel Monitor:
   The basis for SU4.1 references use of the failed fuel monitor to provide indication of fuel clad integrity. The failed fuel monitor at PNP has been abandoned-in-place, and therefore, an alternate method of detection has been developed for SU4.1. Affected EAL: SU4.1 Fuel Clad Degradation

   This deviation is appropriate for PNP as it provides equivalent protection of the public health and safety in the absence of an installed fuel element failure radiation monitor. Additionally, the setpoints selected are high enough to be discriminatory, yet low enough to allow escalation to an Alert in the Fission Product Barrier Degradation category. Therefore, use of these radiation monitors does not decrease the effectiveness of the EAL scheme.

Overview of Significant Differences

Significant differences identified in the matrices include the following:

1. EALs associated with plant equipment not on site were eliminated. This difference is primarily associated with perimeter monitoring and real time dose assessment. This affects EALs RU1.4, RU1.5, RA1.4, RA1.5, RS1.3 and RG1.3.

2. PNP uses two indications for reactor vessel inventory indication (containment sump "OR" primary system drain tank). The "OR" statement was used instead of the NEI wording "and" because there would not be spontaneous indication in both locations. This affects EALs CU2.2, CA1.2, CS1.1, and CG1.1.

3. PNP used bus indication in the classification for loss of offsite power. The PNP wording focuses the classification on the loss of offsite power capability to the essential busses. This affects EALs CU3.1, CA3.1, SU1.1, SS1.1, and SG1.1.

4. PNP is a CE plant and does not use critical safety function status trees, (CSFSTs). Therefore, in accordance with the guidance in NEI 99-01, Revision 4, PNP does not have EALs for CSFSTs under the fuel cladding, primary coolant system, and containment barriers in the Fission Product Barrier Matrix. The loss of the any of the three FPB are addressed under the other loss and/or potential loss indicators.

5. VISIBLE DAMAGE was added as the criteria for determining the effect of the accident conditions for ISFSI. Absent specific criteria in the SARs for the two cask designs used at PNP, a visual assessment is used to determine if there is potential degradation of the fuel cask confinement boundary.

6. PNP is a PWR, therefore, BWR-specific EALs and basis information were removed.

Minor differences identified in the matrices include the following:

Additional minor wording changes have been identified n the generic differences section, as well as in the justification section, for the applicable EAL. These differences do not alter the meaning or intent of the EALs.

Generic Differences
Justification Matrix

EAL Numbering Scheme - PNP utilized a scheme that is different than the NEI 99-01 Rev 4 document. The scheme utilizes different letters as a human performance improvement tool. EALS are described by source S – System Malfunction, H – Hazards and Other, F – Fission Product Barrier, R – Abnormal Radiation (Radiological), C – Cold Shutdown System Malfunctions and E – ISFSI Events. This avoided confusion with the classification designator of U – Unusual Event, A – Alert, S – Site Area Emergency, and G – General Emergency.

Generic Difference 1.

99-01 IC	99-01 EAL #	PNP EAL Number
AU1	1	RU1.1
AU1	2	RU1.2
AU1	3	RU1.3
AU1	4	N/A
AU1	5	N/A
AU2	1	RU2.1
AU2	2	RU2.2
AA1	1	RA1.1
AA1	2	RA1.2
AA1	3	RA1.3
AA1	4	N/A
AA1	5	N/A
AA2	1	RA2.1
AA2	2	RA2.2
AA3	1	RA3.1
AA3	2	RA3.2
AS1	1	RS1.1
AS1	2	RS1.2
AS1	3	N/A
AS1	4	RS1.3
AG1	1	RG1.1
AG1	2	RG1.2
AG1	3	N/A
AG1	4	RG1.3
CU1	1	CU1.1
CU1	2	CU1.2
CU2	1	CU2.1
CU2	2	CU2.2
CU3	1	CU3.1
CU4	1	CU4.1
CU4	2	CU4.2
CU5	1	CU5.1
CU5	2	CU5.2
CU6	1	CU6.1
CU6	2	CU6.2
CU7	1	CU7.1
CU8	1	N/A
CU8	2	CU8.1
CA1	1	CA1.1
CA1	2	CA1.2
CA2	1	CA2.1
CA2	2	CA2.2
CA3	1	CA3.1

Generic Differences
Justification Matrix

99-01 IC	99-01 EAL #	PNP EAL Number
CA4	1	CA4.1
CA4	2	CA4.2
CA4	3	CA4.3
CS1	1	CS1.1
CS1	2	CS1.2
CS2	1	CS2.1
CS2	2	CS2.2
CG1	1	CG1.1
D-AU1	1	N/A
D-AU1	2	N/A
D-AU2	1	N/A
D-SU1	1	N/A
D-SU1	2	N/A
D-HU1	1	N/A
D-HU2	1	N/A
D-HU3	1	N/A
D-HU3	2	N/A
D-HU3	3	N/A
D-HU3	4	N/A
D-HU3	5	N/A
D-HU3	6	N/A
D-HU3	7	N/A
D-HU3	8	N/A
D-AA1	1	N/A
D-AA1	2	N/A
D-AA2	1	N/A
D-AA2	2	N/A
D-HA1	1	N/A
D-HA2	1	N/A
E-HU1	1	EU1.1
E-HU1	2	EU1.2
E-HU1	3	EU1.3
E-HU2	1	EU2.1
FU1		FU1
FA1		FA1
FS1		FS1
FG1		FG1
Fuel Cladding P-Loss - 1		Fuel Cladding P-Loss - 1
Fuel Cladding P-Loss - 2		Fuel Cladding P-Loss - 2
Fuel Cladding P-Loss - 3		Fuel Cladding P-Loss - 3
Fuel Cladding P-Loss - 4		Fuel Cladding P-Loss - 4
Fuel Cladding Loss - 1		Fuel Cladding Loss - 1
Fuel Cladding Loss - 2		Fuel Cladding Loss - 2
Fuel Cladding Loss - 3		Fuel Cladding Loss - 3
Fuel Cladding Loss - 4		Fuel Cladding Loss - 4
Fuel Cladding Loss - 1		Fuel Cladding Loss - 1
Fuel Cladding Loss - 5		Fuel Cladding Loss - 5
Fuel Cladding Loss - 6		Fuel Cladding Loss - 6
RCS P-Loss -1		RCS P-Loss -1
RCS P-Loss -2		RCS P-Loss -2
RCS P-Loss -3		RCS P-Loss -3
RCS P-Loss -4		RCS P-Loss -4

Generic Differences
Justification Matrix

99-01 IC	99-01 EAL #	PNP EAL Number
RCS Loss - 1		PCS Loss - 1
RCS Loss - 2		PCS Loss - 2
RCS Loss - 3		PCS Loss - 3
RCS Loss – 4		PCS Loss - 4
Containment P-Loss - 1		Containment P-Loss - 1
Containment P-Loss - 2		Containment P-Loss - 2
Containment P-Loss - 3		Containment P-Loss - 3
Containment P-Loss – 4		Containment P-Loss – 4
Containment P-Loss – 5		Containment P-Loss – 5
Containment P-Loss – 6		Containment P-Loss – 6
Containment P-Loss – 7		Containment P-Loss – 7
Containment P-Loss - 8		Containment P-Loss - 8
Containment Loss - 1		Containment Loss - 1
Containment Loss – 2		Containment Loss – 2
Containment Loss – 3		Containment Loss – 3
Containment Loss – 4		Containment Loss – 4
Containment Loss – 5		Containment Loss – 5
Containment Loss – 6		Containment Loss – 6
Containment Loss - 7		Containment Loss - 7
HU1	1	HU1.1
HU1	2	HU1.2
HU1	3	HU1.3
HU1	4	HU1.4
HU1	5	HU1.5
HU1	6	HU1.6
HU1	7	HU1.7
HU2	1	HU2.1
HU3	1	HU3.1
HU3	2	HU3.2
HU4	1	HU4.1
HU4	2	HU4.2
HU5	1	HU5.1
HA1	1	HA1.1
HA1	2	HA1.2
HA1	3	HA1.3
HA1	4	HA1.4
HA1	5	HA1.5
HA1	6	HA1.6
HA2	1	HA2.1
HA3	1	HA3.1
HA3	2	HA3.2
HA4	1	HA4.1
HA4	2	HA4.2
HA5	1	HA5.1
HA6	1	HA6.1
HS1	1	HS1.1
HS1	2	HS1.2
HS2	1	HS2.1
HS3	1	HS3.1
HG1	1	HG1.1
HG2	1	HG2.1
SU1	1	SU1.1

Generic Differences
Justification Matrix

99-01 IC	99-01 EAL #	PNP EAL Number
SU2	1	SU2.1
SU3	1	SU3.1
SU4	1	SU4.1
SU4	2	SU4.2
SU5	1	SU5.1
SU5	2	SU5.2
SU6	1	SU6.1
SU6	2	SU6.2
SU8	1	N/A
SU8	2	SU8.1
SA2	1	SA2.1
SA4	1	SA4.1
SA5	1	SA5.1
SS1	1	SS1.1
SS2	1	SS2.1
SS3	1	SS3.1
SS4	1	SS4.1
SS6	1	SS6.1
SG1	1	SG1.1
SG2	1	SG2.1

Generic Difference 2.

NEI Wording	PNP Wording
Increase	Rise
Site specific	The word "increase" has been replaced with "rise" consistent with approved communications terminology. The words 'increase' and 'decrease' are not used because they are easily misunderstood.
Difference	The word "increase" has been replaced with "rise" in accordance with PNP Administrative Procedure 10.51, "Writer's Guideline for Procedures." Administrative Procedure 10.51 directs that words that sound alike but have opposite meanings to not be used. (Formatting change only)

Generic Difference 3.

NEI Wording	PNP Wording
Decrease	Lower(ing)
Site specific	The word "decrease" has been replaced with "lower" or "lowering" to be consistent with approved communications terminology. The words 'increase' and 'decrease' are not used because they are easily misunderstood.
Difference	The word "decrease" has been replaced with "lower" or "lowering" in accordance with PNP Administrative Procedure 10.51, "Writer's Guideline for Procedures." Administrative Procedure 10.51 directs that words that sound alike but have opposite meanings to not be used. (Formatting change only)

Generic Difference 4.

NEI IC Wording	PNP IC Wording
Notification of Unusual Event	Unusual Event

Site specific	The NEI classification "Notification of Unusual Event" has been changed to "Unusual Event" for consistency with the commonly understood title given in the current plant EALs. In addition, Unusual Event is consistent with the terminology used by the State of Michigan.
Difference	The NEI classification "Notification of Unusual Event" has been changed to "Unusual Event" for consistency with the commonly understood title given in the current plant EALs.

Generic Difference 5.

NEI IC Wording	PNP IC Wording
one or more	any
Site specific	The NEI guidance uses both "one or more" and "any" within the document. To implement a consistent approach throughout the EALs, reduce EAL user reading burden, and, thereby increase the potential for timely and accurate emergency classifications, the standard NMC template chose to use "any" throughout the EALs.
Difference	By standard English language definition, "one or more" is equivalent to "any"; therefore this change results in a difference that does not change the intent of the associated EALs.

Generic Difference 6.

NEI IC Wording	PNP IC Wording
Unplanned	UNPLANNED
Site specific	The word "UNPLANNED" is indicated in upper case letters because it has a specific definition associated with its use.
Difference	In accordance with section 5.0 "Definitions" defined terms are to be shown in upper case letters. Placing defined terms in upper case letters is in accordance with the NEI-99-01 rev. 4 guidance. (Formatting change only)

Generic Difference 7.

NEI IC Wording	PNP IC Wording
Greater Than or >	GREATER THAN
Site specific	The word "GREATER THAN" is indicated in upper case letters to be consistent throughout the PNP EALs.
Difference	NEI 99-01 rev. 4 used the words "GREATER THAN" in various formats of upper case text, lower case text, and as a symbol. For PNP to apply appropriate emphasis and consistency the word GREATER THAN has been used in upper case text throughout. In addition, PNP Administrative Procedure 10.51, "Writer's Guideline for Procedures" directs that use of the symbol ">" should be avoided in procedure text because these symbols can be easily misunderstood. (Formatting change only)

Generic Difference 8.

NEI IC Wording	PNP IC Wording

Generic Differences
Justification Matrix

Greater Than or Equal to or ≥	GREATER THAN OR EQUAL TO
Site specific	The word "GREATER THAN OR EQUAL TO" is indicated in capital letters to be consistent throughout the PNP EALs.
Difference	NEI 99-01 rev. 4 used the words "GREATER THAN OR EQUAL TO" in various formats of upper case text, lower case text, and symbols. For PNP to apply appropriate emphasis and consistency the word GREATER THAN OR EQUAL TO has been used in upper case text throughout. In addition, PNP Administrative Procedure 10.51, "Writer's Guideline for Procedures" directs that use of the symbol "≥" should be avoided in procedure text because these symbols can be easily misunderstood. (Formatting change only)

Generic Difference 9.

NEI IC Wording	PNP IC Wording
Less Than or <	LESS THAN
Site specific	The word "LESS THAN" is indicated in upper case letters to be consistent throughout the PNP EALs.
Difference	NEI 99-01 rev. 4 used the words "LESS THAN" in various formats of upper case text, lower case text, and symbols. For PNP to apply appropriate emphasis and consistency the word LESS THAN has been used in upper case text throughout. In addition, PNP Administrative Procedure 10.51, "Writer's Guideline for Procedures" directs that use of the symbol "<" should be avoided in procedure text because these symbols can be easily misunderstood. (Formatting change only)

Generic Difference 10.

NEI IC Wording	PNP IC Wording
Or	OR
Site specific	The word "OR" is indicated in upper case letters when used as a logical operator to ensure that all conditions are clear to the user and to be consistent throughout the PNP EALs.
Difference	PNP Administrative Procedure 10.51, "Writer's Guideline for Procedures" directs that use of the word "OR" be in upper case text to ensure that all conditions are clear to the user. (Formatting change only)

Generic Difference 11.

NEI IC Wording	PNP IC Wording
Scram	trip
Site specific	The term "scram" was replaced with "trip" consistent with PWR terminology.
Difference	The term "scram" was replaced with "trip" consistent with PWR terminology.

Generic Difference 12.

Generic Differences
Justification Matrix

NEI IC Wording	PNP IC Wording
and	AND
Site specific	The word "AND" is indicated in upper case letters when used as a logical operator to ensure that all conditions are clear to the user.
Difference	PNP Administrative Procedure 10.51, "Writer's Guideline for Procedures" directs that use of the word "AND" be in upper case text to ensure that all conditions are clear to the user. (Formatting change only)

Generic Difference 13.

NEI IC Wording	PNP IC Wording
RPV	Reactor Vessel
Site specific	"Reactor Vessel" was used in place of "RPV" to match site procedure verbiage and PWR terminology
Difference	"Reactor Vessel" was used in place of "RPV" to match site procedure verbiage and PWR terminology

Generic Difference 14.

NEI IC Wording	PNP IC Wording
either	EITHER
Site specific	The word "EITHER" is indicated in upper case letters when used as a logical operator to ensure that all conditions are clear to the user.
Difference	PNP Administrative Procedure 10.51, "Writer's Guideline for Procedures" directs that use of logical operators be in upper case text to ensure that all conditions are clear to the user. (Formatting change only)

Generic Difference 15.

NEI IC Wording	PNP IC Wording
Not	NOT
Site specific	The word "NOT" is indicated in upper case letters when used as a logical operator to ensure that all conditions are clear to the user.
Difference	PNP Administrative Procedure 10.51, "Writer's Guideline for Procedures" directs that use of the word "NOT" be in upper case text to ensure that all conditions are clear to the user. (Formatting change only)

Generic Difference 16.

NEI IC Wording	PNP IC Wording
RCS	PCS

Site specific	"PCS" was used in place of "RCS" reactor coolant system acronym to match plant specific terminology
Difference	"PCS" was used in place of "RCS" to match plant specific terminology

Generic Difference 17.

NEI IC Wording	PNP IC Wording
mR, mR/hr	mRem, mRem/hr
Site specific	"mRem" and "mRem/hr" were used in place of "mR" and "mR/hr so as not to be misinterpreted with the term "mRad."
Difference	This change was made to ensure user clarity.

Abnormal Rad Levels / Radiological Effluent Justifications

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
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.	RU1	Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Offsite Dose Calculation Manual (ODCM) Limits for 60 Minutes or Longer.
Mode App.	All		All
Difference	• The Offsite Dose Calculation Manual (ODCM) gives the site-specific limits for gaseous and liquid releases. • Generic Difference 1.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	VALID reading on any effluent monitor that exceeds two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer	RU1.1	Valid reading on effluent monitor RIA-1049 "Liquid Radwaste Discharge Monitor" that exceeds two times the alarm setpoint* established by a current radioactivity discharge permit for 60 minutes or longer. * with waste discharge <u>not</u> isolated
Mode App.	All		All
Site specific	• None		
Difference	• Reference 1, sheets 18 and 19 identify the available monitors. RIA-1049 "Liquid Radwaste Discharge Monitor" replaced the NEI wording "any effluent monitor" since RIA-1049 is the only monitor used for direct discharge to the environment using a radioactivity discharge permit. The alarm setpoint for RIA-1049 is variable and is set in accordance with the ODCM based on the planned release. Using two times the alarm setpoint is consistent with the methodology used in NEI 99-01 and will warn of a release that is not in compliance with the ODCM. The footnote was inserted to avoid declaration if the release is terminated. • Generic Difference 1.		
Deviation	• None		

Abnormal Rad Levels / Radiological Effluent Justifications

NEI EAL#	NEI IC Wording	PNP EAL#(s)	PNP IC Wording
2	VALID reading on one or more of the following radiation monitors that exceeds the reading shown for 60 minutes or longer: (site-specific list)	RU1.2	VALID reading on any of the following radiation monitors that exceeds the reading shown for 60 minutes or longer: Stack Normal Range Monitor (RIA-2326) 3.2E+5 cpm S/G Blowdown Monitor (RIA-0707) 2 X High Alarm* Service Water Monitor (RIA-0833) 2 X High Alarm* Turbine Bldg. Sumps Monitor (RIA-5211) 2 X High Alarm* * with waste discharge <u>not</u> isolated
Mode App.	All		All
Site specific	• Reference 1, sheets 18 and 19 identify the available monitors. The list includes those radiation/effluent monitors, which do not routinely have a radioactivity discharge permit prepared. Reference 1, Table 7 provides the basis for the limits with the exception of RIA-2326. Reference 3, Table 1 provides the basis for the RIA-2326 limit. The limit for RIA-2326 corresponds to two times the ODCM limit. However, the "Alert" alarm on RIA-2326 is set one decade lower. Consequently, the value listed in RU1.2 is 20 times the "Alert" alarm. This meets the intent of NEI 99-01 since the operators will receive sufficient warning of the condition and the limit is based on two times the ODCM limit.		
Difference	• Generic Difference 1.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
3	Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates, with a release duration of 60 minutes or longer, in excess of two times (site-specific technical specifications)	RU1.3	Confirmed sample analysis for gaseous or liquid releases indicates concentrations or release rates, with a release duration of 60 minutes or longer, in excess of two times the ODCM limit.
Site specific	• The ODCM gives the site-specific limits for gaseous and liquid releases.		
Difference	• Generic Difference 1. • The NEI word "analyses" has been replaced with "analysis." This is a grammatical change to allow the subject to agree with the verb "indicates."		
Deviation	• None		

Abnormal Rad Levels / Radiological Effluent Justifications

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
4	VALID reading on perimeter radiation monitoring system greater than 0.10 mR/hr above normal background sustained for 60 minutes or longer [for sites having telemetered perimeter monitors]	N/A	N/A
Site specific	• N/A		
Difference	• Deleted NEI 99-01 example EAL#4 because plant is not equipped with perimeter radiation monitoring.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
5	VALID indication on automatic real-time dose assessment capability greater than (site-specific value) for 60 minutes or longer [for sites having such capacity]	N/A	N/A
Site specific	• N/A		
Difference	• Deleted NEI 99-01 example EAL #5 because plant is not equipped with automatic real-time dose assessment.		
Deviation	• None		

RU1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Replaced Technical Specifications with Offsite Dose Calculation Manual.	1. The ODCM contains the site-specific limits for radiological effluents.
2. Deleted references to EALs #4 and #5.	2. Palisades does not have perimeter radiation monitoring or automatic real-time dose assessment equipment.
Difference	1, 2 The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Abnormal Rad Levels / Radiological Effluent Justifications

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
AU2	Unexpected Increase in Plant Radiation	RU2	Unexpected Rise in Plant Radiation
Mode App.	All		All
Difference	• Generic Difference 1.		
Deviation	• None		

Abnormal Rad Levels / Radiological Effluent Justifications

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	a. VALID (site-specific) indication of uncontrolled water level decrease in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water. AND b. Unplanned VALID (site-specific) Direct Area Radiation Monitor reading increases	RU2.1	VALID indication of uncontrolled water level lowering to LESS THAN 646 ft elevation in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water **AND** UNPLANNED VALID Area Radiation Monitor reading rise on any of the following: Vent Monitor Fuel Handling Area (RIA-5712) Spent Fuel Pool Area Radiation Monitors (RIA-5709 or RIA-2313) Refueling Containment High Radiation (CHR) Monitors (RIA-2316 or RIA-2317)
Site specific	• The minimum allowable water level in the Spent Fuel Pool (SFP) and Refueling Cavity threshold is the Low Spent Fuel Pool water level, alarmed in the Control Room (annunciator EK-1309) at 646 ft elevation or 35 ft above the bottom of the pool. • RIA-5712, RIA-5709, RIA-2313, RIA-2316, and RIA-2317 are the local SFP and Reactor Cavity area monitors required to be in service during fuel moves that have indication available in the control room. References 4 and 5 were used to develop the list of monitors.		
Difference	• The NEI term "Direct" has been deleted because the monitors used at PNP to assess this threshold are commonly referred to as Area Radiation Monitors. • The plant EAL specifically names the monitors applicable to this EAL. Five monitors are provided in the EAL although the NEI indicates only one monitor. Any monitor is used to classify the event because they are available in the control room to aid in classifying the event occurring in the SFP area and the use of any monitor allows for failure of a single monitor. • Generic Differences 1, 3, 5, 6		
Deviation	• None		

Abnormal Rad Levels / Radiological Effluent Justifications

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2	Unplanned VALID Direct Area Radiation Monitor readings increases by a factor of 1000 over normal* levels. *Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.	RU2.2	Any UNPLANNED VALID Area Radiation Monitor reading rises by a factor of 1000 over normal* levels. *Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.
Site specific	• None		
Difference	• The NEI term "Direct" has been deleted because the monitors used at PNP to assess this EAL are commonly referred to as Area Radiation Monitors. • "Any" has been added to clarify the fact that the threshold is met if there is a rise one or more of the indicated readings. • The word "UNPLANNED" is a defined term and is indicated in capital letters. • Generic Differences 1, 2.		
Deviation	• None		

RU2 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Added explanation of specific monitors used. Deleted the duplicate reference to definition of 'Normal' radiation levels.	1. Makes the document more useful for implementer/reviewer.
Difference	1 The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Abnormal Rad Levels / Radiological Effluent Justifications

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
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	RA1	Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Offsite Dose Calculation Manual (ODCM) Limits for 15 Minutes or Longer
Mode App.	All		All
Difference	• The Offsite Dose Calculation Manual (ODCM) gives the site-specific limits for gaseous and liquid releases. • Generic Difference 1.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	VALID reading on any effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer	RA1.1	VALID reading on effluent monitor RIA-1049* "Liquid Radwaste Discharge Monitor" that exceeds 200 times the alarm setpoint* established by a current radioactivity discharge permit for 15 minutes or longer. * with waste discharge <u>not</u> isolated
Site specific	• None		
Difference	• Reference 1, sheets 18 and 19 identify the available monitors. RIA-1049 "Liquid Radwaste Discharge Monitor" replaced the NEI wording "any effluent monitor" since RIA-1049 is the only monitor used for direct discharge to the environment using a radioactivity discharge permit. The alarm setpoint for RIA-1049 is variable and is set in accordance with the ODCM based on the planned release. Using 200 times the alarm setpoint is consistent with the methodology used in NEI 99-01 and will warn of a release that is not in compliance with the ODCM. Reaching the alarm setpoint on RIA-1049 will terminate the release. The footnote was inserted to avoid declaration if the release is terminated. • Generic Difference 1.		
Deviation	• None		

Abnormal Rad Levels / Radiological Effluent Justifications

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2	VALID reading on one or more of the following radiation monitors that exceeds the reading shown for 15 minutes or longer: (site-specific list)	RA1.2	VALID reading on any of the following radiation monitors that exceeds the reading shown for 15 minutes or longer: Stack Normal Range Monitor (RIA-2326) 1.3E+6 cpm Stack High Range Monitor (RIA-2327) 1.5 mRem/hr S/G Blowdown Monitor (RIA-0707) 200 X High Alarm* Service Water Monitor (RIA-0833) 200 X High Alarm* Turbine Bldg. Sumps Monitor (RIA-5211) 200 X High Alarm* * with waste discharge <u>not</u> isolated
Site specific	• Inserted list of radiation/effluent monitors that monitor effluent pathways which do not have a radioactivity discharge permit. The effluent pathways of other monitors are routed through the Stack. Reference 1, sheets 18 and 19 identify the available monitors. The list includes those radiation/effluent monitors, which do not routinely have a radioactivity discharge permit prepared. Reference 1, Table 7 provides the basis for the limits on RIA-0707, RIA-0833, and RIA-5211.		
Difference	• Generic Differences 1, 5		
Deviation	• Reference 3, Table 1 provides the basis for the RIA-2326 and RIA-2327 limits. The limits for RIA-2326 and RIA-2337 are well below the 200X the ODCM limit. These limits were selected to ensure adequate separation from the limits in RS1. The Site Area Emergency limits are based on EPA guidance and uses adverse meteorology while the ODCM values used in RA1.2 use average meteorology. The Alert limits correspond to the High alarm on RIA-2326, which is also the point at which the RGEM monitors switch from low range to high range. While the limits for RIA-2326 and RIA-2327 deviate from the specific guidance in NEI 99-01, they are consistent with the intent to provide a graduated escalation from the UE to the General Emergency.		

Abnormal Rad Levels / Radiological Effluent Justifications

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
3	Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates, with a release duration of 15 minutes or longer, in excess of 200 times (site-specific technical specifications)	RA1.3	Confirmed sample analysis for gaseous or liquid release indicates concentrations or release rates, with a release duration of 15 minutes or longer, in excess of 200 times ODCM limit.
Site specific	• The Offsite Dose Calculation Manual (ODCM) gives the limits for gaseous and liquid releases.		
Difference	• The NEI word "analyses" has been replaced with "analysis." This is a grammatical change to allow the subject to agree with the verb "indicates." • Generic Difference 1.		
Deviation	None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
4	VALID reading on perimeter radiation monitoring system greater than 10.0 mR/hr above normal background sustained for 15 minutes or longer [for sites having telemetered perimeter monitors]	N/A	N/A
Site specific	• None		
Difference	• Deleted NEI 99-01 Example EAL #4 because the plant is not equipped with perimeter radiation monitoring.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
5	VALID indication on automatic real-time dose assessment capability greater than (site-specific value) for 15 minutes or longer [for sites having such capability]	N/A	N/A
Site specific	• None		
Difference	• Deleted NEI 99-01 Example EAL #5 because the plant is not equipped with perimeter radiation monitoring		
Deviation	• None		

Abnormal Rad Levels / Radiological Effluent Justifications

RA1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Replaced Technical Specifications with Offsite Dose Calculation Manual.	1. The ODCM contains the site-specific limits for radiological effluents.
2. Deleted references to EALs #4 and #5.	2. Palisades does not have perimeter radiation monitoring or automatic real-time dose assessment equipment.
Difference	1, 2 The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Abnormal Rad Levels / Radiological Effluent Justifications

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
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	RA2	Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel
Mode App.	All		All
Difference	• Generic Difference 1.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	A VALID (site-specific) alarm or reading on one or more of the following radiation monitors: (site-specific monitors) Refuel Floor Area Radiation Monitor Fuel Handling Building Ventilation Monitor Refueling Bridge Area Radiation Monitor	RA2.1	A VALID alarm or reading on any of the following radiation monitors: Vent Monitor Fuel Handling Area RIA-5712 1E+4 cpm Spent Fuel Pool Area Radiation Monitors RIA-5709 or RIA-2313 15 mRem/hr Refueling Containment High Radiation (CHR) Monitors RIA-2316 or RIA-2317 80 mRem/hr above background
Site specific	• RIA-5712, RIA-5709, RIA-2313, RIA-2316, and RIA-2317 are the local SFP and Reactor Cavity area monitors required to be in service during fuel moves that have indication available in the control room. References 3 and 4 were used to develop the list of monitors. • Reference 1, pages 32 and 36 and reference 2, section 5.8, provide the basis of the alarm values.		
Difference	• Generic Differences 1, 5.		
Deviation	• None		

Abnormal Rad Levels / Radiological Effluent Justifications

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2	Water level less than (site-specific) feet for the reactor refueling cavity, spent fuel pool and fuel transfer canal that will result in irradiated fuel uncovering	RA2.2	Water level less than 636 ft. 9 in. elevation for the reactor refueling cavity, spent fuel pool and fuel transfer canal that will result in fuel uncovering.
Site specific	• Reference 5, sheet 2, provides the basis for the water level. The water level limit corresponds to the top of fuel when located in the fuel handling equipment at the upper limit of travel.		
Difference	• Generic Difference 1.		
Deviation	• None		

RA2 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Reworded some areas.	1. Minor editorial changes were made to eliminate example information and correct references.
Difference	1 The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Abnormal Rad Levels / Radiological Effluent Justifications

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
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	RA3	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
Mode App.	All		All
Difference	• Generic Differences 1, 2.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	VALID (site-specific) radiation monitor readings GREATER THAN 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions: (Site-specific) list	RA3.1	VALID radiation monitor readings GREATER THAN 15 mRem/hr in areas requiring continuous occupancy to maintain plant safety functions: Control Room (RIA-2310) **OR** Central Alarm Station (RIA-2304)
Site specific	• Reference 2, attachment 3 identifies the available radiation monitors. The Control Room (RIA-2310 is located immediately outside the Control Room) and the Central Alarm Station (RIA-2304 is located in the hallway near the Central Alarm Station) are site-specific areas requiring continuous occupancy to maintain plant safety functions.		
Difference	• Generic Difference 1, 17.		
Deviation	• None		

Abnormal Rad Levels / Radiological Effluent Justifications

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2	VALID (site-specific) radiation monitor readings GREATER THAN <site-specific> values in areas requiring infrequent access to maintain plant safety functions. (Site-specific) list	RA3.2	Any VALID radiation monitor reading GREATER THAN 3 R/hr in areas requiring infrequent access to maintain plant safety functions (Table H-1). TABLE H-1 PLANT VITAL AREAS • Containment Structure • Auxiliary Building • Turbine Building • Screenhouse
Site specific	• The basis of the 3 R/hr value is as follows: The PNP annual administrative personnel exposure limit is 2 R/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 R without exceeding the legal 10CFR20 annual exposure limit of 5 R and thus the need for emergency exposure authorization. Assuming that an activity required to be performed in the plant would require a 60-minute stay time in that area an area exposure rate of 3 R/hr would not unduly impede access to areas necessary for safe plant shutdown. • Added Table H-1, Plant VITAL AREAS, as the list of applicable areas for this EAL. Table H-1 was developed based on reference 3, attachment 38. This attachment is derived from the Appendix R safe shutdown analysis and lists those areas considered vital for safe shutdown.		
Difference	• "Any" has been added to clarify the fact that the threshold is met if there is a rise on one or more of the indicated readings. • Generic Difference 1, 17.		
Deviation	• None		

RA3 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Inserted information on determination of radiation level for impeding safe shutdown activities and table including safe shutdown areas.	1. Makes the document more useful for implementer/reviewer.
Difference	1 The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Abnormal Rad Levels / Radiological Effluent Justifications

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
AS1	Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mR TEDE or 500 mR Thyroid CDE for the Actual or Projected Duration of the Release	RS1	Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the Actual or Projected Duration of the Release
Mode App.	All		All
Difference	• Generic Difference 1, 17.		
Deviation	• None		

Abnormal Rad Levels / Radiological Effluent Justifications

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	VALID reading on one or more of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer: (site-specific list) NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL #2 instead of EAL #1.While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.	RS1.1	NOTE: If dose assessment results are available at the time of declaration, the classification should be based on RS1.2 instead of RS1.1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated. VALID reading on any of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer: Stack High Range Monitor (RIA-2327) 5.2 mRem/hr Main Steam Line Monitors (RIA-2323/RIA-2324) 1.70E+3 cpm Stack High Range Effluent Monitor (RIA-2328)* 0.64 mRem/hr Atmospheric Dump Valve High (ADV) Range Effluent Monitors (RIA-2328)* 0.57 mRem/hr *Note RIA-2328 is a backup monitor which can be physically directed for monitoring the ADVs or Stack
Site specific	• Inserted list of the available radiation/effluent monitors. Reference 1, Table 7 provides the basis for the limits on RIA-2327, RIA-2323/RIA-2324, and RIA-2328. As indicated in the note, RIA-2328 is a single monitor that can be aimed at either the plant stack or the atmospheric dump valves. Separate limits are established based on which path is monitored.		
Difference	• Generic Differences 1, 5, 17. • Moved the "NOTE" to the beginning of the EAL as a human factors consideration. Placing notes prior to the applicable steps is consistent with PNP procedure formats. • Replaced "EAL#2" and "EAL#1" with "RG1.2" and "RG1.1."		
Deviation	• None		

Abnormal Rad Levels / Radiological Effluent Justifications

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2	Dose assessment using actual meteorology indicates doses greater than 100 mR TEDE or 500 mR thyroid CDE at or beyond the site boundary	RS1.2	Dose assessment using actual meteorology indicates doses GREATER THAN 100 mRem TEDE or 500 mRem CDE at or beyond the site boundary.
Site specific	• None		
Difference	• Generic Differences 1, 5, 7, 17.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
3	A VALID reading sustained for 15 minutes or longer on perimeter radiation monitoring system greater than 100 mR/hr. [for sites having telemetered perimeter monitors]	N/A	N/A
Site specific	• None		
Difference	• Deleted NEI 99-01 Example EAL #3 because the plant is not equipped with perimeter radiation monitoring.		
Deviation	• None		

Abnormal Rad Levels / Radiological Effluent Justifications

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
4	Field survey results indicate closed window dose rates exceeding 100 mR/hr expected to continue for more than one hour; or analyses of field survey samples indicate thyroid CDE of 500 mR for one hour of inhalation, at or beyond the site boundary	RS1.3	Field survey results indicate closed window dose rates exceeding 100 mRem/hr expected to continue for more than one hour at or beyond the site boundary. **OR** Analyses of field survey samples indicate thyroid CDE of 500 mRem for one hour of inhalation, at or beyond the site boundary
Site specific	• None		
Difference	• "mRad" and "mRem" were used instead of "mR" based on nomenclature from the PNP dose code. "mRad" is used for closed window dose rates. "mRem" is used in reference to CDE. • Added wording "at or beyond the site boundary" before the "OR" to complete the logic of the first criteria. • Generic Differences 1, 10, 17.		
Deviation	• None		

RS1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Deleted references to EAL #3.	1. Palisades does not have perimeter radiation monitoring or automatic real-time dose assessment equipment.
Difference	1 The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Abnormal Rad Levels / Radiological Effluent Justifications

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
AG1	Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mR TEDE or 5000 mR Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology	RG1	Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology
Mode App.	All	All	
Difference	• Generic Difference 1, 17.		
Deviation	• None		

Abnormal Rad Levels / Radiological Effluent Justifications

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	VALID reading on one or more of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer: (site-specific list) NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL #2 instead of EAL #1.While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.	RG1.1	NOTE: If dose assessment results are available at the time of declaration, the classification should be based on RG1.2 instead of RG1.1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated /- completed in order to determine if the classification should be subsequently escalated. VALID reading on any of the following radiation monitors that exceeds or is expected to exceed the reading shown for 15 minutes or longer. Stack High Range Monitor (RIA-2327) 52 mRem/hr Main Steam Line Monitors (RIA-2323/RIA-2324) 1.70E+4 cpm Stack High Range Effluent Monitor (RIA-2328)* 6.4 mRem/hr Atmospheric Dump Valve High Range Effluent Monitors (RIA-2328)* 5.7 mRem/hr *Note RIA-2328 is a backup monitor which can be physically directed for monitoring the ADVs or Stack
Site specific	• Inserted list of the available radiation/effluent monitors. Reference 1, Table 7 provides the basis for the limits on RIA-2327, RIA-2323/RIA-2324, and RIA-2328. As indicated in the note, RIA-2328 is a single monitor that can be aimed at either the plant stack or the atmospheric dump valves. Separate limits are established based on which path is monitored.		
Difference	• Generic Differences 1, 5, 17. • Moved the "NOTE" to the beginning of the EAL as a human factors consideration. Placing notes prior to the applicable steps is consistent with PNP procedure formats. • Replaced "EAL#2" and "EAL#1" with "RG1.2" and "RG1.1."		
Deviation	• None		

Abnormal Rad Levels / Radiological Effluent Justifications

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2	Dose assessment using actual meteorology indicates doses greater than 1000 mR TEDE or 5000 mR thyroid CDE at or beyond the site boundary	RG1.2	Dose assessment using actual meteorology indicates doses GREATER THAN 1000 mRem TEDE or 5000 mRem thyroid CDE at or beyond the site boundary
Site specific	• None		
Difference	• Generic Differences 1, 7, 17.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
3	A VALID reading sustained for 15 minutes or longer on perimeter radiation monitoring system greater than 1000 mR/hr. [for sites having telemetered perimeter monitors]	N/A	N/A
Site specific	• None		
Difference	• Deleted NEI 99-01 Example EAL #3 because the plant is not equipped with perimeter radiation monitoring		
Deviation	• None		

Abnormal Rad Levels / Radiological Effluent Justifications

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
4	Field survey results indicate closed window dose rates exceeding 1000 mR/hr expected to continue for more than one hour; or analyses of field survey samples indicate thyroid CDE of 5000 mR for one hour of inhalation, at or beyond site boundary.	RG1.3	Field survey results indicate closed window dose rates exceeding 1000 mRem/hr expected to continue for more than one hour, at or beyond site boundary. **OR** Analyses of field survey samples indicate thyroid CDE of 5000 mRem for one hour of inhalation, at or beyond site boundary
Site specific	• None		
Difference	• "mRad" and "mRem" were used instead of "mR" based on nomenclature from the PNP dose code. "mRad" is used for closed window dose rates: "mRem" is used in reference to CDE. • Generic Differences 1, 10, 17.		
Deviation	• None		

RG1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Deleted references to EAL #3.	1. Palisades does not have perimeter radiation monitoring or automatic real-time dose assessment equipment.
Difference	1 The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Cold Shutdown Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
CU1	RCS Leakage.	CU1	PCS Leakage.
Mode App.	Cold Shutdown		Cold Shutdown
Difference	• Generic Difference Number 16		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	Unidentified or pressure boundary leakage greater than 10 gpm.	CU1.1.	Unidentified or pressure boundary leakage GREATER THAN 10 gpm.
Site specific	• None		
Difference	• Generic Difference Number 1, 7		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2.	Identified leakage greater than 25 gpm.	CU1.2.	Identified leakage GREATER THAN 25 gpm.
Site specific	• None		
Difference	• Generic Difference Number 1, 7		
Deviation	• None		

Cold Shutdown Justification Matrix

CU1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
• Deleted parenthetical examples of installed instrumentation and NEI Appendix C reference.	• Improved user interpretation.
Difference	• Minor deletions to enhance basis readability.
Deviations	• None

Cold Shutdown Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
CU2	UNPLANNED Loss of RCS Inventory with Irradiated Fuel in the RPV.	CU2	UNPLANNED Loss of PCS Inventory with Irradiated Fuel in the Reactor Vessel.
Mode App.	Refueling		Refueling
Difference	• Generic Difference Number 13, 16		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	UNPLANNED RCS level decrease below the RPV flange for ≥ 15 minutes.	CU2.1.	UNPLANNED PCS level lowering below Reactor Vessel flange for GREATER THAN OR EQUAL TO 15 minutes.
Site specific	• None		
Difference	• Generic Difference Number 1, 8, 13, 16		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2.	a. Loss of RPV inventory as indicated by unexplained {site-specific} sump and tank level increase **AND** b. RPV level cannot be monitored	CU2.2.	Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump level rise OR Primary System Drain Tank level rise. **AND** Reactor Vessel level cannot be monitored
Site specific	• "Containment Sump" and "Primary System Drain Tank" are the site specific indications for loss of Reactor Vessel level inventory		
Difference	• Generic Difference Number 1, 10, 13 • Replaced "and" with "or" as a rise in either Containment Sump or Primary System Drain Tank level will indicate a loss of inventory.		

Cold Shutdown Justification Matrix

Deviation	• None

CU2 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. The PNP specific basis includes additional information related to methods used to determine a loss of PCS inventory and describes the symptoms associated with a loss of PCS inventory.	1. The additional basis information supports the determination of a loss of PCS inventory specified in the EALs.
2. Deleted reference to NEI 99-01 Appendix C.	2. Improved user interpretation.
Difference	1. Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EALs.
	2. Minor deletion to enhance basis readability.
Deviations	• None

Cold Shutdown Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
CU3	Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes	CU3	Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Difference	• Generic Difference Number 7		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	a. Loss of power to (site-specific) transformers for greater than 15 minutes. **AND** b. At least (site-specific) emergency generators are supplying power to emergency busses.	CU3.1.	Loss of all offsite power to Vital 2400 VAC busses 1C and 1D for GREATER THAN 15 minutes. **AND** At least one emergency diesel generator is supplying power to either Vital 2400 VAC bus 1C or 1D.
Site specific	• "Vital 2400 VAC busses 1C and 1D" are the site specific essential busses • "one" is the site-specific number of emergency "diesel" generators needed to power an emergency bus.		
Difference	• Generic Difference Number 1, 7 • The NEI example EAL condition "Loss of power to (site-specific) transformers for greater than 15 minutes" has been changed to "Loss of all offsite power to Vital 2400 VAC busses 1C and 1D for GREATER THAN 15 min." Adding the words "all offsite" to the EAL focuses the classification on the loss of offsite power to the essential busses. Additionally, the words "all offsite" brings information from the basis into the EAL and align the EAL directly to the IC. The PNP wording focuses the classification on the loss of offsite power capability rather than the status of one or more transformers that may or may not be capable of powering the essential buses (e.g. in the event of a fault condition between the transformer and the bus). This simplifies the EAL wording and concisely meets the intent of the NEI IC CU3. The combination of changing transformers to busses, and adding the "all offsite" to the EAL was done to provide the user all the information necessary within the EAL to accurately classify the event. • "either Vital 2400 VAC bus 1C or 1D" added for user clarity.		
Deviation	• None		

Cold Shutdown Justification Matrix

CU3 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. The PNP specific basis includes additional information related to the power supplies for the Vital 2400 VAC busses.	1. The additional basis information supports the determination of a loss of offsite power specified in the EAL.
2. Deleted reference to companion unit AC cross-tie capability.	2. PNP is a single unit plant.

Difference	1. Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EALs.
	2. Deleted reference to companion unit AC cross-tie capability as PNP is a single unit plant.
Deviations	• None

Cold Shutdown Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
CU4	UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the RPV	CU4	UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Difference	• Generic Difference Number 13		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	An UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit	CU4.1.	An UNPLANNED event results in PCS temperature exceeding the Technical Specification cold shutdown temperature limit of 200 degrees F
Site specific	• None		
Difference	• Generic Difference Number 1, 16 • "of 200 °F" added to specify the Technical Specification Cold Shutdown temperature limit.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2.	Loss of all RCS temperature and RPV level indication for > 15 minutes	CU4.2.	Loss of all PCS temperature and Reactor Vessel level indication for GREATER THAN 15 minutes.
Site specific	• None		
Difference	• Generic Difference Number 1, 7, 13, 16		
Deviation	• None		

Cold Shutdown Justification Matrix

CU4 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Revised the wording for escalation to an Alert via CA4.	1. Corrects an internal event escalation inconsistency within the NEI basis document where CU4 basis referenced "30 minutes with CONTAINMENT CLOSURE NOT established" but CA4 EAL states "20 minutes with CONTAINMENT CLOSURE established".
2. Deleted reference to NEI 99-01 Appendix C.	2. Improved user interpretation.
3. Corrected typo: "of" to "or"	3. NEI guidance stated cold shutdown **of** refueling and it should have stated "or refueling."

Difference	1. Revised the discussion for escalation to Alert via CA4.
	2. Minor deletion to enhance basis readability.
	3. Minor correction of typographical error.
Deviations	• None

Cold Shutdown Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
CU5	Fuel Clad Degradation	CU5	Fuel Cladding Degradation
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Difference	• None		
Deviation	• None		

Cold Shutdown Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	(Site-specific) radiation monitor readings indicating fuel clad degradation greater than Technical Specification allowable limits	CU5.1.	Any of the following radiation monitors with a VALID PPC urgent alarm indicating fuel clad degradation GREATER THAN Technical Specification allowable limits. Fuel Handling Area Monitor No. 1 - RIA-2316 Fuel Handling Area Monitor No. 2 - RIA-2317 Containment Isolation High Radiation Monitor - RIA-1805 Containment Isolation High Radiation Monitor - RIA-1806 Containment Isolation High Radiation Monitor - RIA-1807 Containment Isolation High Radiation Monitor - RIA-1808
Site specific	• The Containment Isolation High Radiation and Fuel Handling Area Monitors in high alarm are the "site-specific" fuel clad degradation indications. • For the Containment Isolation High Radiation monitors a special test was conducted that verified that with approximately 1% failed fuel the Containment Isolation High Radiation monitors would indicate 4.5 Rem/hour. The PPC urgent alarm is set at one tenth of that value, 0.45 Rem/hour, indicating fuel clad degradation. For the Fuel Handling Area Radiation monitors a calculation was performed that verified that with approximately 1.0 µCi/gm activity dispersed in the flooded Reactor Cavity during refueling operations the Fuel Handling Area Radiation monitors would indicate 4.8 Rem/hour. Due to the low inventory turnover between the PCS and the Reactor Cavity, PCS activity is assumed to be at least an order of magnitude greater than the Reactor Cavity when the Reactor Cavity activity is at 1.0 µCi/gm. Therefore the PPC urgent alarm is set at 0.48 Rem/hour, one tenth of the calculated reading, indicating fuel clad degradation. This deviation is appropriate for PNP as it provides equivalent protection of the public health and safety in the absence of an installed fuel element failure radiation monitor. Additionally, the setpoints selected are high enough to be discriminatory yet low enough to allow escalation to an Alert in the Fission Product Barrier Degradation category. Therefore use of these radiation monitors does not decrease the effectiveness of the EAL scheme.		
Difference	• Generic Difference Number 1, 7 • PNP uses "with a VALID PPC urgent alarm" as the radiation monitor "reading" due to having these alarm setpoints at a value that indicates fuel clad degradation, the need to provide the operators with an active prompt versus a passive reading from an instrument that is not readily accessed and to align this syntax with that used for other radiation monitor readings in the Abnormal Radiation EALs.		
Deviation	• None		

Cold Shutdown Justification Matrix

Cold Shutdown Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2.	(Site-specific) coolant sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits.	CU5.2.	Coolant sample activity GREATER THAN 1.0 µCi/gm indicating fuel clad degradation.
Site specific	• Less than 1.0 µCi/gm dose equivalent I-131 is the site-specific Technical Specification allowable limit.		
Difference	• Generic Difference Number 1, 7 • "Technical Specification allowable limits" – was deleted as it duplicates the site specific Technical Specification value already listed in the EAL		
Deviation	• None		

Cold Shutdown Justification Matrix

CU5 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. The PNP specific basis includes additional information related to the radiation monitors alarms that would prompt operations to request a PCS sample and activity analysis.	1. PNP has no radiation monitor that could definitively indicate fuel-clad degradation therefore a prompt has been provided to confirm or deny fuel clad degradation.
2. The PNP specific basis includes additional information related to the appropriateness of the coolant sample Technical Specification as a limit for this EAL.	2. The additional basis information supports the determination of fuel cladding degradation specified in the EAL.
Difference	1. & 2. • Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EAL.
Deviations	• None

Cold Shutdown Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
CU6	UNPLANNED Loss of All Onsite or Offsite Communications Capabilities	CU6	UNPLANNED Loss of All Onsite or Offsite Communications Capabilities.
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	Loss of all (site-specific list) onsite communications capability affecting the ability to perform routine operations.	CU6.1.	Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations. **Table C-1 Onsite Communications Systems** • Telephone system • Onsite/offsite radio system • Public address system
Site specific	• Table C-1 lists site-specific equipment used for onsite communications. Additional communications systems listed in Reference #1 for this IC/EAL (i.e. an intercommunications system and the sound-powered phones) would not support routine operations and are not considered appropriate for this EAL.		
Difference	• Generic Difference Number 1		
Deviation	• None		

Cold Shutdown Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2.	Loss of all (site-specific list) offsite communications capability	CU6.2.	Loss of all Table C-2 offsite communications capability. **Table C-2 Offsite Communications Systems** • Telephone system • Power failure phones • FTS phone system • Satellite phone
Site specific	• Table C-2 lists site-specific equipment used for offsite communications. Additional communications systems listed in Reference #1, #2 and #3 for this IC/EAL (e.g. System Controller radio system, on-site/off-site radio system, cell phones, etc) are not considered appropriate for this EAL as these systems grant no assurance that communications with offsite agencies could be established.		
Difference	• Generic Difference Number 1		
Deviation	• None		

CU6 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. The PNP specific basis includes the site-specific Table C-1 and C-2 references related to a loss of communications.	1. The additional basis information supports the determination of a loss of communications specified in the EALs.
Difference	1. Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EALs.
Deviations	• None

Cold Shutdown Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
CU7	UNPLANNED Loss of Required DC Power for Greater than 15 Minutes	CU7	UNPLANNED Loss of Required DC Power for GREATER THAN 15 minutes.
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Difference	• Generic Difference Number 7		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	a. UNPLANNED Loss of Vital DC power to required DC busses based on (site-specific) bus voltage indications. **AND** b. Failure to restore power to at least one required DC bus within 15 minutes from the time of loss.	CU7.1.	UNPLANNED Loss of vital DC power to required DC busses based on bus voltage indications LESS THAN 105 VDC. **AND** Failure to restore power to at least one required DC bus within 15 minutes from the time of loss.
Site specific	• LESS THAN 105 VDC is the site-specific value for a loss of Vital DC.		
Difference	• Generic Difference Number 1, 9		
Deviation	• None		

CU7 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. The PNP specific basis includes additional information related to Station Battery capabilities.	1. The additional basis information supports the determination of the loss of DC power specified in the EAL.
Difference	1. Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EAL.
Deviations	• None

Cold Shutdown Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
CU8	Inadvertent Criticality.	CU8	Inadvertent Criticality.
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	An UNPLANNED extended positive period observed on nuclear instrumentation.	(N/A)	(Not used)
Site specific	• None		
Difference	• PNP is a PWR and has no period nuclear instrumentation.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2.	An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.	CU8.1.	An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.
Site specific	• None		
Difference	• Generic Difference Number 1		
Deviation	• None		

Cold Shutdown Justification Matrix

CU8 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Deleted reference to period monitors.	1. PNP is a PWR and has no period nuclear instrumentation.
Difference	1. BWR-specific EAL not used.
Deviations	• None

Cold Shutdown Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
CA1	Loss of RCS Inventory.	CA1	Loss of PCS Inventory.
Mode App.	Cold Shutdown		Cold Shutdown
Difference	• Generic Difference Number 16		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	Loss of RCS inventory as indicated by RPV level less than {site-specific level}. (low-low ECCS actuation setpoint) (BWR) (bottom ID of the RCS loop) (PWR)	CA1.1.	Loss of PCS inventory as indicated by Reactor Vessel level LESS THAN 617 ft. 0 in. elevation.
Site specific	• 617 ft. 0 in. (See deviation below)		
Difference	• Generic Difference Number 1, 9, 13, 16 • Low-low ECCS actuation setpoint is not used at PNP		
Deviation	• When Reactor Vessel water level lowers to 616 ft. 5.5 in. elevation, the bottom of the PCS hot leg is uncovered. Reactor Vessel Level Transmitter LT-0105 and Reactor Hot Leg Level Transmitter LT-0106 provide the closest readily available indication of this level. Both transmitters will indicate approximately 0% at a level of 617 ft. 0 in. Refer to Addendum #1 for additional justification for this deviation related to PNPs' ability to properly classify and escalate this event using installed plant equipment.		

Cold Shutdown Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2.	a. Loss of RCS inventory as indicated by unexplained {site-specific} sump and tank level increase **AND** b. RCS level cannot be monitored for > 15 minutes	CA1.2.	Loss of PCS inventory as indicated by unexplained Containment Sump OR Primary System Drain Tank level rise **AND** Reactor Vessel level cannot be monitored for GREATER THAN 15 minutes
Site specific	• Generic Difference Number 1, 7, 16 • "Containment Sump OR Primary System Drain Tank" level rise are the site-specific indications for loss of Reactor Vessel level inventory		
Difference	• Replaced "and" with "or" as a rise in either Containment Sump or Primary System Drain Tank level will indicate a loss of inventory. • Replaced "RCS" with "Reactor Vessel" for consistency with other EALs that describe monitoring Reactor Vessel level.		
Deviation	• None		

CA1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Deleted reference to BWR ECCS actuation and NEI 99-01 Appendix C reference.	1. Improved user interpretation.
2. Information added on site specific level used in CA1.1.	2. PNP site specific details added to basis.
Difference	1. Minor deletions to enhance basis readability.
	2. Added site specific information in basis to support EAL with regard to level for CA1.1.
Deviations	• See specific EAL justification (CA1.1)

Cold Shutdown Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
CA2	Loss of RPV Inventory with Irradiated Fuel in the RPV	CA2	Loss of Reactor Vessel Inventory with Irradiated Fuel in the Reactor Vessel
Mode App.	Refueling		Refueling
Difference	• Generic Difference Number 13		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	Loss of RPV inventory as indicated by RPV level less than {site-specific level}. (low-low ECCS actuation setpoint) (BWR) (bottom ID of the RCS loop) (PWR)	CA2.1.	Loss of Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 617 ft. 0 in. elevation.
Site specific	• 617 ft. 0 in. (See deviation below)		
Difference	• Generic Difference Number 1, 9, 13 • Low-low ECCS actuation setpoint is not used at PNP		
Deviation	• When Reactor Vessel water level lowers to 616 ft. 5.5 in. elevation, the bottom of the PCS hot leg is uncovered. Reactor Vessel Level Transmitter LT-0105 and Reactor Hot Leg Level Transmitter LT-0106 provide the closest readily available indication of this level. Both transmitters will indicate approximately 0% at a level of 617 ft. 0 in. Refer to Addendum #1 for additional justification for this deviation related to PNPs' ability to properly classify and escalate this event using installed plant equipment.		

Cold Shutdown Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2.	a. Loss of RPV inventory as indicated by unexplained {site-specific} sump and tank level increase **AND** b. RPV level cannot be monitored for > 15 minutes	CA2.2.	Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump OR Primary System Drain Tank level rise **AND** Reactor Vessel level cannot be monitored for GREATER THAN 15 minutes
Site specific	• Generic Difference Number 1, 7, 10, 13 • "Containment Sump OR Primary System Drain Tank" level rise are the site-specific indications for loss of Reactor Vessel level inventory		
Difference	• Replaced "and" with "or" as a rise in either Containment Sump or Primary System Drain Tank level will indicate a loss of inventory.		
Deviation	• None		

CA2 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Deleted reference to BWR ECCS actuation and NEI 99-01 Appendix C reference.	1. Improved user interpretation.
2. Added information to support site specific levels	2. Add site specific information for PNP.
Difference	1. Minor deletions to enhance basis readability.
Deviations	• See specific EAL justification (CA2.1)

Cold Shutdown Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
CA3	Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses.	CA3	Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses.
Mode App.	Cold Shutdown, Refueling, Defueled		Cold Shutdown, Refueling, Defueled
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	a. Loss of power to (site-specific) transformers. **AND** b. Failure of (site-specific) emergency generators to supply power to emergency busses. **AND** c. Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.	CA3.1.	Loss of all offsite power to both Vital 2400 VAC busses 1C and 1D **AND** Failure of both emergency diesel generators to supply power to Vital 2400 VAC busses. **AND** Failure to restore power to at least one Vital 2400 VAC bus within 15 minutes from the time of loss of both offsite and onsite AC power.
Site specific	• "Vital 2400 VAC buses 1C and 1D" are the site-specific essential busses. • Emergency "diesel" generator is site specific nomenclature used at PNP. • "both" is site-specific for the emergency "diesel" generators .		
Difference	• Generic Difference Number 1 • The NEI example EAL condition "Loss of power to (site-specific) transformers" has been changed to "Loss of all offsite power to both Vital 2400 VAC busses 1C and 1D" Adding the words "all offsite" to the EAL focuses the classification on the loss of offsite power to the essential busses. Additionally, the words "all offsite" brings information from the basis into the EAL and align the EAL directly to the IC. The PNP wording focuses the classification on the loss of offsite power capability rather than the status of one or more transformers that may or may not be capable of powering the essential buses (e.g. in the event of a fault condition between the transformer and the bus). This simplifies the EAL wording and concisely meets the intent of the NEI IC CA3. The combination of changing transformers to busses, and adding the "all offsite" to the EAL was done to provide the user all the information necessary within the EAL to accurately classify the event. • Replaced "emergency" with "Vital 2400 VAC" for user clarity.		

Cold Shutdown Justification Matrix

Deviation	• None

Cold Shutdown Justification Matrix

CA3 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. The PNP specific basis includes additional information related to the power supplies for the Vital 2400 VAC busses.	1. The additional basis information supports the determination of a loss of offsite power specified in the EAL.

Difference	1. Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EAL.
Deviations	• None

Cold Shutdown Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
CA4	Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the RPV	CA4	Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the Reactor Vessel.
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Difference	• Generic Difference Number 13		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	With CONTAINMENT CLOSURE <u>and</u> RCS integrity <u>not</u> established an UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit.	CA4.1.	With CONTAINMENT CLOSURE <u>and</u> PCS integrity <u>not</u> established an UNPLANNED event results in PCS temperature exceeding the Technical Specification cold shutdown temperature limit of 200 degrees F.
Site specific	• None		
Difference	• Generic Difference Number 1, 16 • "of 200 degrees F" was added to specify the Technical Specification Cold Shutdown temperature limit.		
Deviation	• None		

Cold Shutdown Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2.	With CONTAINMENT CLOSURE established and RCS integrity not established or RCS inventory reduced an UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit for greater than 20 minutes[1]. [1]Note: if an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced then this EAL is not applicable.	CA4.2.	With CONTAINMENT CLOSURE established and PCS integrity not established or PCS inventory reduced an UNPLANNED event results in PCS temperature exceeding the Technical Specification cold shutdown temperature limit of 200 degrees F for GREATER THAN 20 minutes[1]. [1]Note: if a PCS heat removal system is in operation within this time frame and PCS temperature is being reduced then this EAL is not applicable.
Site specific	• None		
Difference	• Generic Difference Number 1, 7, 16 • "of 200°F" was added to specify the Technical Specification Cold Shutdown temperature limit. • Brought Note into EAL from basis. • Replaced "an" with "a" in Note 1 for proper grammar use with "PCS".		
Deviation	• None		

Cold Shutdown Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
3.	An UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit for greater than 60 minutes[1] or results in an RCS pressure increase of greater than {site-specific} psig. [1]Note: if an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced then this EAL is not applicable.	CA4.3.	An UNPLANNED event results in PCS temperature exceeding the Technical Specification cold shutdown temperature limit of 200 degrees F for GREATER THAN 60 minutes[1] OR results in a PCS pressure rise of GREATER THAN 10 psig. [1]Note: if a PCS heat removal system is in operation within this time frame and PCS temperature is being reduced then this EAL is not applicable.
Site specific	• 10 psig is the site-specific value for the PCS pressure rise.		
Difference	• Generic Difference Number 1, 7, 10, 16 • "of 200°F" was added to specify the Technical Specification Cold Shutdown temperature limit. • Brought Note into EAL from basis. • Replaced "an" with "a" in Note 1 for proper grammar use with "PCS".		
Deviation	• None		

CA4 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. The PNP specific basis includes additional information related to PCS pressure indication that replaced the NEI 99-01 discussion of "Control Board instrumentation".	1. The additional basis information supports the determination the inability to maintain cold shutdown specified in the EALs.
2. Deleted reference to NEI 99-01 Appendix C.	2. Improved user interpretation.
Difference	1. Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EAL.
	2. Minor deletion to enhance basis readability.
Deviations	• None

Cold Shutdown Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
CS1	Loss of RPV Inventory Affecting Core Decay Heat Removal Capability	CS1	Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability
Mode App.	Cold Shutdown		Cold Shutdown
Difference	• Generic Difference Number 13		
Deviation	• None		

Cold Shutdown Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	With CONTAINMENT CLOSURE not established: a. RPV inventory as indicated by RPV level less than {site-specific level} (6" below the low-low ECCS actuation setpoint) (BWR) (6" below the bottom ID of the RCS loop) (PWR) **OR** b. RPV level cannot be monitored for > 30 minutes with a loss of RPV inventory as indicated by unexplained {site-specific} sump and tank level increase	CS1.1.	With CONTAINMENT CLOSURE not established: a. Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 616 ft. 6 in. elevation OR b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by unexplained Containment Sump OR Primary System Drain Tank level rise.
Site specific	• "616 ft. 6 in. elevation" is the site-specific value for 6" below the bottom ID of the PCS hot leg loop. • "Containment Sump or Primary System Drain Tank" level rise are the site-specific indications for loss of Reactor Vessel level inventory.		
Difference	• Generic Difference Number 1, 7, 9, 10, 13 • Replaced "and" with "or" as a rise in either Containment Sump or Primary System Drain Tank level will indicate a loss of inventory. • Low-low ECCS actuation setpoint is not used at PNP.		
Deviation	• When Reactor Vessel water level lowers to 615 ft. 11.5 in. elevation, level is 6 in. below the bottom ID of the PCS Hot Leg. RVLMS is unable to distinguish 6 in. below the bottom ID of the PCS loop penetration. The closest indication of this level is provided by RVLMS UGS Region Sensor #7 red light on, ~40 in. above the fuel alignment plate at 616 ft. 6 in. el.. This indication allows clear escalation from CA1.1 and maintains the 6 in. difference in PCS elevation between the Alert and Site Area Emergency classification prescribed by NEI 99-01. Refer to Addendum #1 for additional justification for this deviation related to PNPs' ability to properly classify and escalate this event using installed plant equipment.		

Cold Shutdown Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2.	With CONTAINMENT CLOSURE established a. RPV inventory as indicated by RPV level less than TOAF **OR** b. RPV level cannot be monitored for > 30 minutes with a loss of RPV inventory as indicated by either: • Unexplained {site-specific} sump and tank level increase • Erratic Source Range Monitor Indication	CS1.2.	With CONTAINMENT CLOSURE established: a. Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 614 ft. 0 in. elevation OR b. Reactor Vessel level cannot be monitored for GREATER THAN 30 minutes with a loss of Reactor Vessel inventory as indicated by either: • Unexplained Containment Sump OR Primary System Drain Tank level rise • Erratic Source Range Monitor Indication
Site specific	• "Containment Sump or Primary System Drain Tank" level rise are the site-specific indications for loss of Reactor Vessel level inventory.		
Difference	• Generic Difference Number 1, 7, 9, 10, 13 • Replaced "and" with "or" as a rise in either Containment Sump or Primary System Drain Tank level will indicate a loss of inventory. • Low-low ECCS actuation setpoint is not used at PNP.		
Deviation	• "614 ft. 0 in. el." is ~11 in. above the "TOAF" and is the lowest level indication that can be used to determine Reactor Vessel inventory loss at PNP. Refer to Addendum #1 for additional justification for this deviation related to PNPs' ability to properly classify and escalate this event using installed plant equipment.		

Cold Shutdown Justification Matrix

CS1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. The PNP specific basis includes additional information related to methods used to determine a loss of PCS inventory and describes the symptoms associated with a loss of PCS inventory.	1. The additional basis information supports the determination of a loss of PCS inventory that affects decay heat removal specified in the EALs.
2. Deleted reference to BWR-specific EAL determinations.	2. PNP is a PWR.
3. Deleted reference to NEI 99-01 Appendix C.	3. Improved user interpretation.

Difference	1. Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EALs.
	2. BWR-specific references deleted.
	3. Minor deletion to enhance basis readability.
Deviations	• See specific EAL justifications (CS1.1, CS1.2)

Cold Shutdown Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
CS2	Loss of RPV Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the RPV.	CS2	Loss of Reactor Vessel Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the Reactor Vessel.
Mode App.	Refueling		Refueling
Difference	• Generic Difference Number 13		
Deviation	• None		

Cold Shutdown Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	With CONTAINMENT CLOSURE not established: a. RPV inventory as indicated by RPV level less than {site-specific level} (6" below the low-low ECCS actuation setpoint) (BWR) (6" below the bottom ID of the RCS loop) (PWR) **OR** b. RPV level cannot be monitored with Indication of core uncovery as evidenced by one or more of the following: • Containment High Range Radiation Monitor reading > {site-specific} setpoint • Erratic Source Range Monitor Indication • Other {site-specific} indications	CS2.1.	With CONTAINMENT CLOSURE not established: a. Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 616 ft. 6 in. elevation OR b. Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by any of the following: • Containment High Range Radiation Monitor reading GREATER THAN 40 R/hr • Erratic Source Range Monitor Indication
Site specific	• 40 R/hr is the site-specific value for core uncovery indication from the Containment High Range Monitor reading. • "Other {site-specific} indications" are not used at PNP.		
Difference	• Generic Difference Number 1, 5, 7, 9, 13		
Deviation	• When Reactor Vessel water level lowers to 615 ft. 11.5 in. elevation, level is 6 in. below the bottom ID of the PCS Hot Leg. RVLMS is unable to distinguish 6 in. below the bottom ID of the PCS loop penetration. The closest indication of this level is provided by RVLMS UGS Region Sensor #7 red light on, ~40 in. above the fuel alignment plate at 616 ft. 6 in. el.. This indication allows clear escalation from CA2.1 and maintains the 6 in. difference in PCS elevation between the Alert and Site Area Emergency classification prescribed by NEI 99-01. Refer to Addendum #1 for additional justification for this deviation related to PNPs' ability to properly classify and escalate this event using installed plant equipment.		

Cold Shutdown Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2.	With CONTAINMENT CLOSURE established a. RPV inventory as indicated by RPV level less than TOAF **OR** b. RPV level cannot be monitored with Indication of core uncovery as evidenced by one or more of the following: • Containment High Range Radiation Monitor reading > {site-specific} setpoint • Erratic Source Range Monitor Indication • Other {site-specific} indications	CS2.2.	With CONTAINMENT CLOSURE established: a. Reactor Vessel inventory as indicated by Reactor Vessel level LESS THAN 614 ft. 0 in. elevation OR b. Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by any of the following: • Containment High Range Radiation Monitor reading GREATER THAN 40 R/hr • Erratic Source Range Monitor Indication
Site specific	• "40 R/hr" is the site-specific value for core uncovery indication from the Containment High Range Monitor reading • "Other {site-specific} indications" are not used at PNP.		
Difference	• Generic Difference Number 1, 5, 7, 9, 13		
Deviation	• "614 ft. 0 in. el." is ~11 in. above the "TOAF" and is the lowest level indication that can be used to determine Reactor Vessel inventory loss at PNP. When Reactor Vessel water level drops to 613 ft. 2 in. elevation (Top Of Active Fuel) [Ref. 2], core uncovery is about to occur. The closest indication of this level is provided by RVLMS UGS Region Sensor #8 red light on, ~11 in. above the fuel alignment plate at 614 ft. 0 in. elevation. Refer to Addendum #1 for additional justification for this deviation related to PNPs' ability to properly classify and escalate this event using installed plant equipment.		

Cold Shutdown Justification Matrix

CS2 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. The PNP specific basis includes additional information related to methods used to determine a loss of PCS inventory and describes the symptoms associated with a loss of PCS inventory.	1. The additional basis information supports the determination of a loss of PCS inventory that affects decay heat removal specified in the EALs.
2. Deleted reference to BWR-specific EAL determinations.	2. PNP is a PWR.
3. Deleted reference to NEI 99-01 Appendix C.	3. Improved user interpretation.

Difference	1. Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EALs.
	2. BWR-specific references deleted.
	3. Minor deletion to enhance basis readability.
Deviations	• See specific EAL justifications (CS2.1, CS2.2)

Cold Shutdown Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
CG1	Loss of RPV Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the RPV	CG1	Loss of Reactor Vessel Inventory Affecting Fuel Cladding Integrity with Containment Challenged and Irradiated Fuel in the Reactor Vessel.
Mode App.	Cold Shutdown, Refueling		Cold Shutdown, Refueling
Difference	• Generic Difference Number 13		
Deviation	• None		

Cold Shutdown Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1. and 2. and 3.	1. Loss of RPV inventory as indicated by unexplained {site-specific} sump and tank level increase 2. RPV Level: a. less than TOAF for > 30 minutes **OR** b. cannot be monitored with Indication of core uncovery for > 30 minutes as evidenced by one or more of the following: • Containment High Range Radiation Monitor reading > {site-specific} setpoint • Erratic Source Range Monitor Indication • Other {site-specific} indications 3. {Site-specific} indication of CONTAINMENT challenged as indicated by one or more of the following: • Explosive mixture inside containment • Pressure above {site-specific} value • CONTAINMENT CLOSURE <u>not</u> established • Secondary Containment radiation monitors above {site-specific} value (BWR only)	CG1.1	Loss of Reactor Vessel inventory as indicated by unexplained Containment Sump OR Primary System Drain Tank level rise. **AND** Reactor Vessel Level: a. LESS THAN 614 ft. 0 in. elevation for GREATER THAN 30 minutes OR b. cannot be monitored with indication of core uncovery for GREATER THAN 30 minutes as evidenced by any of the following: • Containment High Range Radiation Monitor reading GREATER THAN 40 R/hr • Erratic Source Range Monitor Indication **AND** Indication of Containment challenged as indicated by any of the following: • Containment hydrogen concentration GREATER THAN OR EQUAL TO 6% • Containment pressure above 55 psig • CONTAINMENT CLOSURE <u>not</u> established
Site specific	• "Containment Sump or Primary System Drain Tank" level rise are the "Other site-specific indications" for loss of Reactor Vessel level inventory. • "10 R/hr" is the site-specific value for core uncovery indication from the Containment High Range Monitor reading. • "Other {site-specific} indications" are not used at PNP. • "6%" is the site-specific CONTAINMENT challenge Containment hydrogen explosive mixture. • "55 psig" is the site-specific CONTAINMENT challenge Containment pressure. • "Secondary Containment radiation monitors above {site-specific} value (BWR only)" is		

	not applicable to PNP.
Difference	• Generic Difference Number 1, 5, 7, 8, 9, 10 • Replaced "and" with "or" as a rise in either Containment Sump or Primary System Drain Tank level will indicate a loss of inventory. • "Explosive mixture inside containment" was replaced with "Containment hydrogen concentration GREATER THAN OR EQUAL TO 6%" for user clarity. • "Pressure above {site-specific} value" was replaced with "Containment pressure above 55 psig" for user clarity.
Deviation	• "614 ft. 0 in. el." is ~11 in. above the "TOAF" and is the lowest level indication that can be used to determine Reactor Vessel inventory loss at PNP. When Reactor Vessel water level drops to 613 ft. 2 in. elevation (Top Of Active Fuel) [Ref. 2], core uncovery is about to occur. The closest indication of this level is provided by RVLMS UGS Region Sensor #8 red light on, ~11 in. above the fuel alignment plate at 614 ft. 0 in. elevation. Refer to Addendum #1 for additional justification for this deviation related to PNPs' ability to properly classify and escalate this event using installed plant equipment.

CG1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. The PNP specific basis includes additional information related to methods used to determine a loss of PCS inventory, fuel clad integrity threats and challenges to Containment.	1. The additional basis information supports the determination of a loss of PCS inventory that affects fuel clad integrity with containment challenged and irradiated fuel in the vessel specified in the EALs.
2. Deleted reference to BWR-specific EAL determinations.	2. PNP is a PWR.
3. Deleted reference to NEI 99-01 Appendix C.	3. Improved user interpretation.
Difference	1. Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EALs.
	2. BWR-specific references deleted.
	3. Minor deletion to enhance basis readability
Deviations	• See specific EAL justification (CG1.1)

ISFSI Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
E-HU1	Damage to loaded cask CONFINEMENT BOUNDARY	EU1	Damage to loaded cask CONFINEMENT BOUNDARY
Mode App.	Not Applicable		Not Applicable
Difference	• Generic Difference 1.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	Natural phenomena events affecting a loaded cask CONFINEMENT BOUNDARY. (site-specific list)	EU1.1	Natural phenomena events affecting a loaded cask CONFINEMENT BOUNDARY as indicated by VISIBLE DAMAGE: • earthquake • tornado • flood • seiche • lightning
Site specific	• The list of natural phenomena was developed from the Safety Analysis Reports (SAR) for the two cask designs used at PNP. Reference 1, Section 2.5 "Protection Against Environmental Condition And Natural Phenomena" was used with Reference 2, Sections 3.2 "Structural and Mechanical Safety Criteria" and Section 8.2 "Accident Analysis" to derive the site specific list.		
Difference	• Generic Difference 1. • VISIBLE DAMAGE was added as the criteria for determining the effect of the natural phenomena. The SARs for the two cask designs used at PNP do not establish any measurable threshold values for determining degradation of the confinement boundary. Absent these specific criteria, a visual assessment of the Transfer Cask, Ventilated Concrete Cask or Horizontal Storage Module is used to determine if there is potential degradation of the fuel cask CONFINEMENT BOUNDARY. This clarifying statement was added to support EAL determination. The additional information does not impact the classification or intent of the associated EAL.		
Deviation	• None		

ISFSI Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2	Accident conditions affecting a loaded cask CONFINEMENT BOUNDARY. (site-specific list)	EU1.2	Accident conditions affecting a loaded cask CONFINEMENT BOUNDARY as indicated by VISIBLE DAMAGE: • tipped over or dropped cask • FIRE • EXPLOSION
Site specific	• The list of accident conditions was developed from the Safety Analysis Reports (SAR) for the two cask designs used at PNP. Reference 1, Section 2.6 "Protection Against Fire And Explosion" was used with Reference 2, Sections 3.2 "Structural and Mechanical Safety Criteria" and Section 8.2 "Accident Analysis" to derive the site-specific list.		
Difference	• Generic Difference 1. • VISIBLE DAMAGE was added as the criteria for determining the effect of the accident conditions. The SARs for the two cask designs used at PNP do not establish any measurable threshold values for determining degradation of the confinement boundary. Absent these specific criteria, a visual assessment of the Transfer Cask, Ventilated Concrete Cask or Horizontal Storage Module is used to determine if there is potential degradation of the fuel cask CONFINEMENT BOUNDARY. This clarifying statement was added to support EAL determination. The additional information does not impact the classification or intent of the associated EAL.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
3	Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask CONFINEMENT BOUNDARY.	EU1.3	Any condition in the opinion of the Emergency Director that indicates loss of loaded fuel storage cask CONFINEMENT BOUNDARY.
Site specific	• None		
Difference	• Generic Difference 1.		
Deviation	• None		

ISFSI Justification Matrix

EU1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Added a discussion on the use of VISIBLE DAMAGE as the criteria for classification under this EAL.	1. VISIBLE DAMAGE was added as the criteria for determining the effect of the natural phenomena. The SARs for the two cask designs used at PNP do not establish any measurable threshold values for determining degradation of the confinement boundary. Absent these specific criteria, a visual assessment of the Transfer Cask, Ventilated Concrete Cask or Horizontal Storage Module is used to determine if there is potential degradation of the fuel cask CONFINEMENT BOUNDARY.
2. Added clarification that classification under this EAL is dependent on damage to the cask not the intensity of the natural phenomena or accident.	3. For the events of concern here, the key issue is whether the resultant damage to or loss of the loaded fuel cask CONFINEMENT BOUNDARY leads to the degradation of the fuel during transfer or storage, or poses an operational safety problem with respect to its removal from storage.
4. The basis discussion on development of the EALs utilizing the SAR was deleted.	4. The discussion on basis development adds no value for the decision makers and may hinder their decision because of extraneous material.
Difference	1, 2, 3, 4 • Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EAL(s).
Deviation	• None

ISFSI Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
E-HU2	Confirmed Security Event with potential loss of level of safety of the ISFSI.	EU2	Confirmed Security Event with potential loss of level of safety of the ISFSI.
Mode App.	Not applicable.		Not applicable
Difference	• Generic Difference 1.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	Security Event as determined from (site-specific) Security Plan and reported by the (site-specific) security shift supervision.	EU2.1	Security Event as determined from the PNP Security Plan and reported by the Security Shift Leader.
Site specific	• The site-specific security plan is the "PNP Security Plan." • The security shift supervision is titled the "Security Shift Leader."		
Difference	• Generic Difference 1.		
Deviation	• None		

EU2 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Added "the PNP."	1. The title of the site-specific contingency plan "the PNP Security Plan" was added to the basis for clarity.
2. Added the title of "Security Shift Leader"	2. The site-specific title of the security shift supervisor, "Security Shift Leader" was added to the basis for clarity.
Difference	1, 2 • The additional information does not impact the classification or intent of the associated EALs.
Deviation	• None

Fission Product Barrier Degradation Justification Matrix

NEI IC Matrix #(s)	NEI IC Matrix Wording	PNP IC Matrix #(s)	PNP IC Matrix Wording
Table 5-F-1	Recognition Category F Fission Product Barrier Degradation INITIATING CONDITION MATRIX	Table F-0	Recognition Category F Fission Product Barrier Degradation INITIATING CONDITION MATRIX
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Power Operation, Hot Standby, Startup, Hot Shutdown
Difference	• Table number designation changed to Table F-0 to eliminate BWR information and differentiate between this table and numbering for Table F-1 • Deleted reference to "(See Sections 3.4 and 3.8)" • Deleted reference to Tables 3 and 4 and replaced with reference to Table F-1 • Changed the word "state" to "states" • These changes are not a deviation because they do not alter the meaning or intent, such that classification of the event could be different between the NEI guidance and the plant EAL. • Generic difference number 4		
Deviation	• None		

NEI Fission Product Barrier Reference Table	NEI Table Wording	PNP Fission Product Barrier Reference Table	PNP Table Wording
Table 5-F-2	Table and associated Basis are applicable to BWRs.	N/A	None
Difference	• Table not applicable to Palisades. Palisades is a PWR.		
Deviation	• None		

Fission Product Barrier Degradation Justification Matrix

NEI Fission Product Barrier Reference Table #(s)	NEI Fission Product Barrier Reference Table Wording	PNP Fission Product Barrier Reference Table #(s)	PNP Fission Product Barrier Reference Table Wording
Table 5-F-4	PWR Emergency Action Level Fission Product Barrier Reference Table Thresholds for LOSS or POTENTIAL LOSS of Barriers*	Table F-1	PNP Emergency Action Level Fission Product Barrier Reference Table Thresholds for LOSS or POTENTIAL LOSS of Barriers*
Site specific	• None		
Difference	• Table number designation changed to Table F-1. The table number was changed due to the elimination of the BWR information, and to differentiate between this table and the numbering for Table F-0. • Replaced the word "for" with the word "or" in the asterisked note. • Deleted the word "Example" from all three barrier EALS. • Replaced PWR with PNP to be plant specific.		
Deviation	• None		

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
FU1	**UNUSUAL EVENT** ANY Loss or ANY Potential Loss of Containment	FU1	**UNUSUAL EVENT** ANY Loss or ANY Potential Loss of Containment
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Power Operation, Hot Standby, Startup, Hot Shutdown
Difference	• Aligned the upper case and lower case text of the NEI 99-01 classification wording to be consistent between Table F-0 and Table F-1.		
Deviation	• None		

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
FA1	**ALERT** ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS	FA1	**ALERT** ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR PCS
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Power Operation, Hot Standby, Startup, Hot Shutdown
Difference	• Generic difference number 16		
Deviation	• None		

Fission Product Barrier Degradation Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
FS1	**SITE AREA EMERGENCY** Loss or Potential Loss of ANY Two Barriers	FS1	**SITE AREA EMERGENCY** Loss or Potential Loss of ANY Two Barriers
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Power Operation, Hot Standby, Startup, Hot Shutdown
Difference	• None		
Deviation	• None		

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
FG1	**GENERAL EMERGENCY** Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier	FG1	**GENERAL EMERGENCY** Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier
Mode App.	Power Operation, Hot Standby, Startup, Hot Shutdown		Power Operation, Hot Standby, Startup, Hot Shutdown
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
Fuel Clad Loss 1	**Critical Safety Function Status** Core-Cooling Red	Fuel Clad Loss 1	**Critical Safety Function Status** Not Applicable
Fuel Clad Potential Loss 1	**Critical Safety Function Status** Core Cooling-Orange OR Heat Sink-Red	Fuel Clad Potential Loss 1	
Site specific	• None		
Difference	• Not applicable per NEI 99-01 Revision 4. PNP is a CE design reactor and as such does not implement CSFSTs. There is no equivalent Core Cooling color designated path conditions. The loss of core cooling fuel clad barrier thresholds are addressed under primary coolant activity level, the CET readings, reactor vessel water level and the containment radiation monitoring for the loss and/or potential loss indicators.		
Deviation	• None		

Fission Product Barrier Degradation Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
Fuel Clad Loss 2	**Primary Coolant Activity Level** Coolant Activity GREATER THAN (site specific) Value	Fuel Clad Loss 2	**Primary Coolant Activity Level** Dose rate value for primary coolant GREATER THAN 1 R/hr
Site specific	• Listed the site-specific dose rate value for primary coolant of GREATER THAN 1 R/hr as specified in reference 6, Emergency Implementing Procedure, EI-7.0, "Emergency Post Accident Sampling and Determination of Fuel Failure Using Dose Rates". This value is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage.		
Difference	• PNP is unable to analyze a primary coolant sample of 300 µci/gm due to high dose rates experienced at the sample panel. Therefore, a dose rate is used in lieu of a sample to definitively indicate fuel clad degradation as allowed by the NEI 99-01 basis. The dose rate value used for the primary coolant is specified in accordance with reference 6, Emergency Implementing Procedure, EI-7.0, "Emergency Post Accident Sampling and Determination of Fuel Failure Using Dose Rates". The sample is taken to measure contact dose rates at the Failed Fuel Survey Point on the PCS sample line. • Generic difference number 7		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
Fuel Clad Loss 3	**Core Exit Thermocouple Readings** GREATER THAN (site-specific) degree F	Fuel Clad Loss 3	**Core Exit Thermocouple Readings** GREATER THAN 1200 degree F
Site specific	• 1200 degrees F is the site-specific core exit temperature as specified in Reference 3, "Generic Accident Management Guidelines" and NEI 99-01 revision 4.		
Difference	• None		
Deviation	• None		

Fission Product Barrier Degradation Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
Fuel Clad Potential Loss 3	**Core Exit Thermocouple Readings** GREATER THAN (site-specific) degree F	Fuel Clad Potential Loss 3	**Core Exit Thermocouple Readings** GREATER THAN 700 degree F
Site specific	• 700 degrees F CET temperature is the highest postulated DBA temperature in accordance with the FSAR and is therefore indicative of a potential fuel clad loss. Per reference 2, "EOP Setpoint Basis" for the minimum PCS subcooling.		
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
Fuel Clad Potential Loss 4	**Reactor Vessel Water Level** Level LESS than (site-specific) value	Fuel Clad Potential Loss 4	**Reactor Vessel Water Level** Level LESS THAN 614 ft. 0 in. elevation
Site specific	• 614 ft. 0 in. elevation corresponds to the lowest indicated liquid level above the active fuel in accordance with the FSAR, Section 7.4.6.3, "Reactor Vessel Level Monitoring System" and Vendor Drawing, "RLI Display Panel". [Ref. 18, 19]		
Difference	• Generic difference number 9		
Deviation	• In accordance with NEI 99-01 rev. 4 the basis information specifies the level in this EAL is to correspond to top of active fuel. RVLMS is the instrumentation used to indicate reactor vessel level. The lowest reactor vessel water level indication available on the RVLMS is 11 inches above the bottom of the fuel alignment plate, (which is approximately the top of the active fuel). As specified in FSAR, Section 7.4.6.3, "Reactor Vessel Level Monitoring System" and Vendor Drawing, "RLI Display Panel". [Ref. 18, 19] • This deviation is appropriate for PNP based on the RVLMS system design as it provides equivalent protection of the public health and safety by ensuring that the potential loss of the fuel clad barrier is determined before the TOAF is uncovered. Within the current EAL scheme the fuel clad EAL for core exit thermocouple temperature "potential loss" is available to support the determination of the "potential loss" of the fuel clad. Refer to Addendum #1 for additional justification for this deviation related to PNPs' ability to properly classify and escalate this event using installed plant equipment.		

Fission Product Barrier Degradation Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
Fuel Clad Loss 5	**Containment Radiation Monitoring** Containment rad monitor reading GREATER THAN (site-specific) R/hr	Fuel Clad Loss 5	**Containment Radiation Monitoring** Containment High Range Radiation Monitor reading GREATER THAN 2000 R/hr as indicated on RIA-2321 and RIA-2322
Site specific	• 2000 R/hr is the site-specific Containment High Range Radiation Monitor reading that has been calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I-131 into the containment atmosphere per Reference 4, Engineering Calculation EA-JLV-04-01.		
Difference	• Inserted site-specific containment radiation monitors RIA-2321 and RIA-2322 utilized in EAL determination, for ease of use.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
Fuel Clad Loss 6	**Other (Site-Specific) Indications** (Site-specific) as applicable	Fuel Clad Loss 6	**Other Indications** Not Applicable
Fuel Clad Potential Loss 6		Fuel Clad Potential Loss 6	
Site specific	• None		
Difference	• Replaced "(Site-specific) as applicable" with "Not applicable" for clarity. • No other applicable site-specific indications of fuel cladding loss and/or potential loss exist.		
Deviation	• None		

Fission Product Barrier Degradation Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
Fuel Clad Loss 7	**Emergency Director Judgment** Any condition in the opinion of the Emergency Director that indicate Loss or Potential Loss of the Fuel Clad Barrier	Fuel Clad Loss 7	**Emergency Director Judgment** Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Fuel Clad Barrier
Fuel Clad Potential Loss 7		Fuel Clad Potential Loss 7	
Site specific	• None		
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
RCS Loss 1	**Critical Safety Function Status** Not Applicable	PCS Loss 1	**Critical Safety Function Status** Not Applicable
RCS Potential Loss 1	**Critical Safety Function Status** RCS Integrity-Red OR Heat Sink-Red	PCS Potential Loss 1	
Site specific	• None		
Difference	• Not applicable per NEI 99-01 Revision 4. PNP is a CE design reactor and as such does not implement CSFSTs. There is no equivalent color designated path conditions. The loss of the PCS barrier threshold is addressed under PCS leak and S/G Tube Rupture for loss and/or potential loss EALs. • Generic difference 16		
Deviation	• None		

Fission Product Barrier Degradation Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
RCS Loss 2	**RCS Leak Rate** GREATER THAN available makeup capacity as indicated by a loss of RCS subcooling	PCS Loss 2	**PCS Leak Rate** GREATER THAN available makeup capacity indicated by PCS subcooling LESS THAN 25 degrees F based on average of qualified CETs
Site specific	• None		
Difference	• PCS subcooling LESS THAN 25 degrees F represents the subcooling margin threshold as specified in Off Normal and Emergency Operating Procedures as measured by the average of the qualified CETS. CETs LESS THAN 25 degrees F provides the subcooling margin threshold that encompasses variations in available makeup capacity. By definition, EOPs define this value as a loss of PCS subcooling. • Generic difference numbers 9, 16		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
RCS Potential Loss 2	**RCS Leak Rate** Unisolable leak exceeding the capacity of one charging pump in the normal charging mode	PCS Potential Loss 2	**PCS Leak Rate** Unisolable leak exceeding 50 gpm
Site specific	• None		
Difference	• The normal operating charging pump is variable speed, with a capacity of 33 to 53 gpm. The other two charging pumps are fixed speed, with a lower capacity of 40 gpm each. PNP has low capacity charging pumps; therefore per NEI guidance a 50 gpm leak rate value is used to indicate the Potential Loss. • Generic difference numbers 9, 16		
Deviation	• None		

Fission Product Barrier Degradation Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
RCS Loss 3	**SG Tube Rupture** SGTR that results in an ECCS (SI) Actuation	PCS Loss 3	**SG Tube Rupture** SGTR that results in an ECCS (SI) Actuation
Site specific	• None		
Difference	• Generic difference number 16		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
RCS Loss 4	**Containment Radiation Monitoring** Containment rad monitor reading GREATER THAN (site-specific) R/hr	PCS Loss 4	**Containment Radiation Monitoring** Containment High Range Radiation Monitor reading GREATER THAN 200 R/hr as indicated on RIA-2321 and RIA-2322
Site specific	• 200 R/hr is the site-specific Containment High Range Radiation Monitor reading that has been calculated assuming the instantaneous release and dispersal of the primary coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within T/S) into the containment atmosphere. Per Reference 4, Engineering Analysis EA-JLV-04-01. • Generic difference number 16		
Difference	• Inserted site-specific containment radiation monitors RIA-2321 and RIA-2322 utilized in EAL determination for ease of use.		
Deviation	• None		

Fission Product Barrier Degradation Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
RCS Loss 5	**Other (Site-Specific) Indications** (Site-specific) as applicable	PCS Loss 5	**Other Indications** Not Applicable
RCS Potential 5		PCS Potential Loss 5	
Site specific	• "(Site-specific) as applicable" with "Not applicable" for clarity • No other applicable site-specific indications of a PCS barrier loss exist.		
Difference	• Generic difference number 16		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
RCS Loss 6	**Emergency Director Judgment** Any condition in the opinion of the Emergency Director that indicate Loss or Potential Loss of the RCS Barrier	PCS Loss 6	**Emergency Director Judgment** Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the PCS Barrier
RCS Potential Loss 6		PCS Potential Loss 6	
Site specific	• None		
Difference	• Generic difference number 16		
Deviation	• None		

Fission Product Barrier Degradation Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
CMT Loss 1	**Critical Safety Function Status** Not Applicable	CMT Loss 1	**Critical Safety Function Status** Not Applicable
CMT Potential Loss 1	**Critical Safety Function Status** Containment-Red	CMT Potential Loss 1	
Site specific	• None		
Difference	• Not applicable per NEI 99-01 Revision 4. PNP is a CE design reactor and as such does not implement CSFSTs. There is no equivalent Containment color designated path condition. Loss of the Containment barrier thresholds are addressed under the other Containment loss and potential loss EALs.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
CMT Loss 2	**Containment Pressure** Rapid unexplained decrease following initial increase **OR** Containment pressure or sump level response not consistent with LOCA conditions	CMT Loss 2	**Containment Pressure** Rapid unexplained lowering following initial rise **OR** Containment pressure or sump level response not consistent with LOCA conditions
Site specific	• None		
Difference	• Generic difference number 2, 3		
Deviation	• None		

Fission Product Barrier Degradation Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
CMT Potential Loss 2	**Containment Pressure** (Site-specific) PSIG and increasing OR Explosive mixture exists **OR** Pressure greater than containment depressurization actuation setpoint with less than one full train of depressurization equipment operating	CMT Potential Loss 2	**Containment Pressure** 55 psig and rising OR Containment hydrogen concentration GREATER THAN OR EQUAL TO 6% OR Containment pressure GREATER THAN 4 psig with LESS THAN one full train of depressurization equipment operating
Site specific	• "55 psig" is the site-specific Containment pressure for potential loss of Containment based on FSAR design pressure.		
Difference	• Replaced the words "Site-Specific" from EAL wording with plant specific containment pressure. • "6%" is the site-specific potential loss of Containment hydrogen explosive mixture based on RG 1.7. • Inserted the 4 psig actuation setpoint of the containment heat removal/depressurization system per reference 17, ARP-8, "Safeguards Safety Injection and Isolation Scheme EK-13" Annunciator 61 and reference 11, EOP-1 "Standard Post-Trip Actions" section 4.0 • Generic difference number 2, 7, 8		
Deviation	• None		

Fission Product Barrier Degradation Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
CMT Potential Loss 3	**Core Exit Themocouple Reading** 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	CMT Potential Loss 3	**Core Exit Thermocouple Reading** Core exit thermocouples in excess of 1200 degrees F and restoration procedures not effective within 15 minutes; OR Core exit thermocouples in excess of 700 degrees F with reactor vessel level below 614 ft. 0 in. elevation and restoration procedures not effective within 15 minutes
Site specific	• None		
Difference	• Capitalized the letter "C" and separated the two statements in EAL for user clarity. • Generic difference number 10		
Deviation	• Deleted the words in the EAL "below top of active fuel" and replaced with "614 ft. 0 in. elevation." The 614 feet 0 inch elevation corresponds to the lowest indicated liquid level above the active fuel to determine core uncovery. RVLMS is the instrumentation used to indicate reactor vessel level. The lowest indication on the RVLMS is 11 inches above the bottom of the fuel alignment plate, which is approximately the top of the active fuel. Refer to Addendum #1 for additional justification for this deviation related to PNPs' ability to properly classify and escalate this event using installed plant equipment.		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
CMT Loss 4	**SG Secondary Side Release with P-to-S Leakage** RUPTURED S/G is also FAULTED outside of containment OR Primary-to-Secondary leakrate greater than 10 gpm with nonisolable steam release from affected S/G to the environment	CMT Loss 4	**SG Secondary Side Release with P-to-S Leakage** RUPTURED S/G is also FAULTED outside of containment OR Primary-to-Secondary leakrate GREATER THAN 10 gpm with nonisolable steam release from affected S/G to the environment
Site specific	• None		
Difference	• Generic difference number 7		
Deviation	• None		

Fission Product Barrier Degradation Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
CMT Loss 5	**CNMT Isolation Valves Status After CNMT Isolation** Valve(s) not closed AND downstream pathway to the environment exists	CMT Loss 5	**CNMT Isolation Valves Status After CNMT Isolation** Containment isolation Valve(s) not closed AND Downstream pathway to the environment exists after containment isolation
Site specific	• None		
Difference	• "Containment isolation valve(s)" wording was used to address the NEI title wording. • Generic difference number 12		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
CMT Potential Loss 6	**Significant Radioactive Inventory in Containment** Containment rad monitor reading GREATER THAN (site-specific) R/hr	CMT Potential Loss 6	**Significant Radioactive Inventory in Containment** Containment High Range Radiation Monitor reading GREATER THAN 20,000 R/hr as indicated on RIA-2321 and RIA-2322
Site specific	• Inserted site-specific value of 20,000 R/hr, which indicates significant fuel damage well in excess of the EALs associated with both loss of Fuel Clad and loss of PCS Barriers as determined by Reference 4, Engineering Analysis EA-JLV-04-01.		
Difference	• Inserted site-specific Containment High Range Radiation Monitors RIA-2321 and RIA-2322 utilized to indicate significant fuel damage for ease of EAL use. • Generic difference number 7		
Deviation	• None		

Fission Product Barrier Degradation Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
CMT Loss 7	**Other (Site-Specific) Indications** (Site-specific) as applicable	CMT Loss 7	**Other Indications** Not Applicable
CMT Potential Loss 7		CMT Potential Loss 7	
Site specific	• Replaced "(Site-specific) as applicable" with "Not applicable" for clarity • No other applicable site-specific indications of a Containment Barrier potential loss exist.		
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
CMT Loss 8	**Emergency Director Judgment** Any condition in the opinion of the Emergency Director that indicate Loss or Potential Loss of the Containment Barrier	CMT Loss 8	**Emergency Director Judgment** Any condition in the opinion of the Emergency Director that indicate Loss or Potential Loss of the Containment Barrier
CMT Potential Loss 8		CMT Potential Loss 8	
Site specific	• None		
Difference	• None		
Deviation	• None		

Fission Product Barrier Degradation Justification Matrix

Fuel Clad Barrier - Loss and/or Potential Loss Basis Justification	
PNP Specific Additions/Deletions	**Justification**
In the fuel clad barrier basis all discussion related to CSFSTs was deleted.	The deleted CSFST basis information is not applicable to PNP, which is a CE design reactor.
1. **Critical Safety Function Status** PNP is a CE design reactor and does not implement CSFSTs therefore the basis information was deleted. **For the remaining fuel clad barrier EALs all discussion in the basis related to CSFSTs was deleted.**	1. **Critical Safety Function Status** The deleted basis information is addressed by the other Fuel Clad loss and/or potential loss EALs. **For the remaining fuel clad barrier EALs the deleted CSFST basis information in not applicable to PNP, which is a CE design reactor.**
2. **Primary Coolant Activity Level** The PNP specific basis includes additional information related to the primary coolant dose rate value used to indicate significant clad damage.	2. **Primary Coolant Activity Level** The additional basis information correlates the specified dose rate value to radioactivity levels that indicate significant clad damage, and thus a value related to the loss of the Fuel Clad Barrier. The 1 R/hr value is obtained from reference 6 procedure EI-7.0, "Emergency Post Accident Sampling and Determination of Fuel Failure Using Dose Rates" Attachment 2 graph by approximating a dose rate value above 1% failed fuel that approaches 5% failed fuel.
3. **Core Exit Thermocouple Readings**. Inserted plant specific information related to the use of the CETs for both the loss and potential loss EALs.	3. **Core Exit Thermocouple Readings** Added site-specific basis information for the specified CET temperatures readings for loss and potential loss EAL classification. • The Loss threshold temperature is provided per reference 3, CEOG Generic Accident Management Guidelines, "Phase 1.0- Initial Diagnosis." • The Potential Loss threshold temperature provided per reference 2, "EOP Setpoint Basis," is the temperature indicative of a potential fuel clad loss.

Fission Product Barrier Degradation Justification Matrix

4. **Reactor Vessel Water Level** Additional information has been included in the basis discussing the lowest indicated level reactor water level on the RVLMS.	4. **Reactor Vessel Water Level** In accordance with NEI 99-01 rev. 4 the basis information specifies the level in this EAL is to correspond to top of active fuel. RVLMS is the instrumentation used to indicate reactor vessel level. The additional basis discusses the RVLMS sensor locations that correlate reactor water level. The lowest reactor vessel water level indication available on the RVLMS is 11 inches above the bottom of the fuel alignment plate, (which is approximately the top of the active fuel). As specified in FSAR, Section 7.4.6.3, "Reactor Vessel Level Monitoring System" and Vendor Drawing, "RLI Display Panel". [Ref. 18, 19] Therefore, the lowest indicated water level elevation of below 614 feet was used in place of the top of the active fuel for EAL determination.
5. **Containment Radiation Monitoring** Inserted the site-specific containment radiation monitor equipment numbers into the basis. Replaced "site specific" with the radiation reading that indicates radiation values related to primary coolant release. Included radiation monitor reading for the high alarm setpoint.	4. **Containment Radiation Monitoring** The equipment numbers for the Containment High Range Radiation Monitor were included to clarify equipment used to determine increased radiation levels in containment. The site-specific radiation value of 2000 R/hr was included as determined by reference 4, Engineering Analysis EA-JLV-04-01. This value indicates a release of reactor coolant with elevated activity indicative of fuel damage. For clarity the radiation monitor high alarm value was included. Deleted the NEI information that was not part of a plant specific EAL basis.
6. **Other Indications** No additional PNP information has been added to the basis.	5. **Other Indications** No other site-specific indications of a fuel cladding loss or potential loss exist.
7. **Emergency Director Judgment** The PNP specific basis includes additional information related to aiding the Emergency Director in determining if conditions exists that indicate the loss or potential loss of the fuel clad barrier.	6. **Emergency Director Judgment** The additional bulleted items in the basis for Emergency Director judgment are an amalgam of bases information from NEI 99-01 revision 4. The first bulleted item comes from the notes on Table 5-F-1 as well as sections 3.9 and 3.10 of the NEI document regarding "imminent" barrier loss. The second bulleted item is from the bases of IC HG1, loss of all AC, regarding degraded barrier monitoring capability that appears appropriate here. The third bulleted item also comes from IC HG2 as well as SG2 (ATWS) regarding the importance of the use of ED judgment to make anticipatory declarations based on FPB monitoring.

Fission Product Barrier Degradation Justification Matrix

Differences	1, 2, 3, 5, 6, 7 • Deleted CSFSTs basis information. • Additional information was added to support EAL determination. • The additional information does not impact the classification or intent of the associated EAL(s).
Deviation	4- As discussed above the value used to determine top of active fuel is based on the lowest reactor water level indication. This deviation is appropriate for PNP based on the RVLMS system design as it provides equivalent protection of the public health and safety by ensuring that the potential loss of the fuel clad barrier is determined before the TOAF is uncovered. Within the current EAL scheme the fuel clad EAL for core exit thermocouple temperature "potential loss" is available to support the determination of the "potential loss" of the fuel clad. Refer to Addendum #1 for additional justification for this deviation related to PNPs' ability to properly classify and escalate this event using installed plant equipment.

PCS Barrier - Loss and/or Potential Loss Basis Justification	
PNP Specific Additions/Deletions	**Justification**
In the PCS barrier basis all discussion related to CSFSTs was deleted.	The deleted CSFST basis information is not applicable to PNP, which is a CE design reactor.
1. **Critical Safety Function Status** PNP is a CE design reactor and does not implement CSFSTs therefore the basis information was deleted **For the remaining PCS barrier EALs all discussion in the basis related to CSFSTs was deleted.**	1. **Critical Safety Function Status** The deleted basis information is addressed by the other PCS Barrier Loss and/or Potential Loss EALs **For the remaining PCS barrier EALs the deleted CSFST basis information is not applicable to PNP, which is a CE design reactor.**
2. **PCS Leak Rate** The PNP specific basis includes additional information related to charging pump capacity to support the 50 gpm flowrate.	2. **PCS Leak Rate** The additional information included in the basis explains the plant specific Chemical and Volume Control System information to support the 50 gpm charging pump flowrate in the EAL. It discusses the charging pump design and normal system operation. The "Potential Loss" EAL is based on the inability to maintain normal liquid inventory within the PCS by normal operation of the Chemical and Volume Control System, which is considered as one charging pump discharging to the charging header.
3. **SG Tube Rupture** Deleted the reference and information related to the Westinghouse Owners Group emergency response guides	3. **SG Tube Rupture** Westinghouse Owners Group emergency response guides are not used at PNP, due PNP a CE design reactor.
4. **Containment Radiation Monitoring** Inserted the site-specific containment radiation monitor equipment numbers into the basis. Replaced "site specific" with the radiation reading that indicates radiation values related to primary coolant release. Included radiation monitor reading for the	4. **Containment Radiation Monitoring** The PNP Containment High Range Radiation Monitors selected support EAL determination, and therefore this EAL has been included as an indication of PCS barrier loss. The containment radiation monitor equipment numbers and their alarm values were included to clarify equipment used to determine

high alarm setpoint. Deleted the NEI basis discussion related to omitting this EAL if an appropriate radiation monitor was not available.	rising radiation levels in containment. Replaced "site specific" with the site radiation value of 200 R/hr as the value that indicates release of primary coolant in containment. This value indicates a release of reactor coolant with elevated activity indicative of fuel damage as determined by reference 4, Engineering Analysis EA-JLV-04-01. For clarity the basis included the selected radiation monitor alarm values. Deleted the NEI information that was not part of a plant specific EAL basis.
5. **Other Indications** No additional PNP information has been added to the basis	5. **Other Indications** No other site-specific indications of a PCS loss or potential loss exist.
6. **Emergency Director Judgment** The PNP specific basis includes additional information related to aiding the Emergency Director in determining if conditions exists that indicate the loss or potential loss of the PCS barrier.	6. **Emergency Director Judgment** The additional bulleted items in the basis for Emergency Director judgment are an amalgam of bases information from NEI 99-01 revision 4. The first bulleted item comes from the notes on Table 5-F-1 as well as sections 3.9 and 3.10 of the NEI document regarding "imminent" barrier loss. The second bulleted item is from the bases of IC HG1, loss of all AC, regarding degraded barrier monitoring capability that appears appropriate here. The third bulleted item also comes from IC HG2 as well as SG2 (ATWS) regarding the importance of the use of ED judgment to make anticipatory declarations based on FPB monitoring.
Differences	1, 2, 3, 4. 5, 6, Deleted the CSFSTs basis information since it is not utilized at PNP. Deleted reference to the Westinghouse Owners Group reference guides, PNP is a CE design plant. Additional information was added to support EAL determination and to input plant specific information. All additional information does not impact the classification or intent of the associated EAL(s).
Deviation	• None

Containment Barrier - Loss and/or Potential Loss Basis Justification	
PNP Specific Additions/Deletions	**Justification**
In the containment barrier basis all discussion related to CSFSTs was deleted. Deleted NEI background information that is not intended for a plant specific EAL basis.	The deleted CSFST basis information in not applicable to PNP, which is a CE design reactor. The deleted NEI background information was replaced with plant specific EAL basis information where appropriate.
1. **Critical Safety Function Status** PNP is a CE design reactor and does not implement CSFSTs therefore the basis information was deleted.	1. **Critical Safety Function Status** The deleted basis information is addressed by the other Containment Barrier Loss and/or Potential Loss EALs

Fission Product Barrier Degradation Justification Matrix

For the remaining Containment barrier EALs all discussion in the basis related to CSFSTs was deleted.	**For the remaining Containment barrier EALs the deleted CSFST basis information is not applicable to PNP, which is a CE design reactor.**
2. **Containment Pressure** Inserted PNP plant specific basis for containment pressure and replaced the generic NEI basis comment to be plant specific.	2. **Containment Pressure** Additional basis information was included to support the lower explosive limit value. In addition, the containment heat removal/depressurization system actuation setpoint was added to the EAL for clarity. The site-specific information was added as required per the NEI basis and for EAL clarity.
3. **Core Exit Thermocouple Readings** Inserted plant specific information related to the use of the CETs. Added information supports the specified CET temperatures reading and relates the indicated reactor vessel level relationship to the top of the active fuel.	3. **Core Exit Thermocouple Readings** Additional basis information supports the specified CET temperatures reading and clarifies the basis for the chosen reactor vessel level for PNP. The NEI basis states that the reactor vessel level chosen should be consistent with the emergency response guides for the facility. This statement was deleted and made plant specific. The inserted basis information relates the use of the lowest RVLMS indication of 11 inches above the bottom of the fuel alignment plate, to the top of the active fuel, which corresponds to a reactor vessel level elevation of below 614' 0" elevation. Therefore, 614' 0" elevation, is the site-specific reactor vessel water level above the active fuel to determine core uncovery.
4. **SG Secondary Side Release with P-to-S Leakage** Added the definitions for a faulted and/or ruptured SG. Deleted the NEI basis discussion related to the development of this EAL and the previous revisions to the NEI document	4. **SG Secondary Side Release with P-to-S Leakage** Faulted and ruptured SG definitions were added for EAL basis clarity. The deleted NEI basis information that was in the basis for establishing the plant specific EAL basis. The deleted information is not required for a plant specific.
5. **CNMT Isolation Valves Status After CNMT Isolation** No changes from the NEI basis.	5. **CNMT Isolation Valves Status After CNMT Isolation** No changes from the NEI basis.
6. **Significant Radioactive Inventory in Containment** Inserted the site-specific containment radiation monitor equipment numbers into the basis. Included radiation monitor reading for the high alarm setpoint. Deleted NEI basis information used to establish the plant specific EAL basis.	6. **Significant Radioactive Inventory in Containment** Plant specific information was inserted as required in the NEI basis. The additional equipment numbers for the Containment High Range Radiation Monitors were included to clarify equipment used to determine rising radiation levels in containment. Replaced "site specific" with the site radiation value of 20,000 R/hr as the value that indicates significant fuel damage well in excess of the EALs associated with both loss of fuel clad and loss of PCS as determined by reference 4, Engineering Analysis EA-JLV-04-01. For clarity the site-specific radiation monitors and their alarm values were included. Deleted the NEI information that was not part of a plant specific EAL basis.
7. **Other Indications** No additional PNP information has been added to the basis.	7. **Other Indications** No other site-specific indications of a containment barrier loss or potential loss exist.

Fission Product Barrier Degradation Justification Matrix

8. **Emergency Director Judgment** The PNP specific basis includes additional information related to aiding the Emergency Director in determining if conditions exists that indicate the loss or potential loss of the fuel clad barrier.	8. **Emergency Director Judgment** The additional bulleted items in the basis for Emergency Director judgment are an amalgam of bases information from NEI 99-01 revision 4. The first bulleted item comes from the notes on Table 5-F-1 as well as sections 3.9 and 3.10 of the NEI document regarding "imminent" barrier loss. The second bulleted item is from the bases of IC HG1, loss of all AC, regarding degraded barrier monitoring capability that appears appropriate here. The third bulleted item also comes from IC HG2 as well as SG2 (ATWS) regarding the importance of the use of ED judgment to make anticipatory declarations based on FPB monitoring.

Differences	1, 2, 4, 6, 7, 8 • Deleted Critical Safety Function Status basis information was deleted since it is not utilized at PNP. • Deleted NEI background information that is not intended for a plant specific EAL basis. • Added plant specific information as appropriate and where required by the NEI basis.
Deviation	3 - As discussed above the value used to determine top of active fuel is based on the lowest reactor water level indication. This deviation is appropriate for PNP based on the RVLMS system design as it provides equivalent protection of the public health and safety by ensuring that the potential loss of the fuel clad barrier is determined before the TOAF is uncovered. Within the current EAL scheme the fuel clad EAL for core exit thermocouple temperature "potential loss" is available to support the determination of the "potential loss" of the fuel clad. Refer to Addendum #1 for additional justification for this deviation related to PNPs' ability to properly classify and escalate this event using installed plant equipment.

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HU1	Natural and Destructive Phenomena Affecting the PROTECTED AREA.	HU1	Natural and Destructive Phenomena Affecting the PROTECTED AREA.
Mode App.	All		All
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	(Site-Specific) method indicates felt earthquake.	HU1.1	Earthquake felt in plant as indicated by: Report by Plant Personnel to on duty Control Room Personnel **AND** Confirmation from the National Earthquake Information Center
Site specific	• Site-specific methods of detecting an earthquake are a report by plant personnel confirmed by the National Earthquake Information Center. PNP has no seismic monitoring equipment that would alarm and warn the control room staff of an earthquake. Reference 2, Section 4.14 "Earthquakes" contains the directions for contacting the National Earthquake Information Center for validation by a reliable source in accordance with NEI 99-01, Rev 4 basis information.		
Difference	• Generic Difference Number1. • EAL wording was modified to allow listing of the methods of detection.		
Deviation	• None		

Hazards Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2	Report by plant personnel of tornado or high winds greater than (site-specific) mph striking within PROTECTED AREA boundary.	HU1.2	Report by plant personnel of tornado or high winds GREATER THAN 95 mph striking within PROTECTED AREA boundary.
Site specific	• The site-specific value of "95 mph" wind speed was inserted to meet the FSAR 100 mph threshold within limitations of site instrumentation. Site instrumentation will only read up to 100 mph. A margin of 5 mph is considered an acceptable value for the instrumentation's accuracy. Reference 3, Section 5.3 "Wind And Tornado Loadings" lists the 100 mph design limit.		
Difference	• Generic Difference Number 1, 7		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
3	Vehicle crash into plant structures or systems within PROTECTED AREA boundary.	HU1.3	Vehicle crash into plant structures or systems within PROTECTED AREA boundary.
Site specific	• None		
Difference	• Generic Difference Number 1		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
4	Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.	HU1.4	Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.
Site specific	• None		
Difference	• Generic Difference Number 1		
Deviation	• None		

Hazards Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
5	Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.	HU1.5	Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.
Site specific	• None		
Difference	• Generic Difference Number 1		
Deviation	• None		

Hazards Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
6	Uncontrolled flooding in (site-specific) areas of the plant that has the potential to affect safety related equipment needed for the current operating mode.	HU1.6	Uncontrolled flooding in the following areas of the plant that has the potential to affect safety related equipment needed for the current operating mode: • Emergency Diesel Generator Rooms • Engineered Safeguards Rooms • Auxiliary Feedwater Pump Room • Switchgear Room 1C • Screen House • Component Cooling Water Pump Room
Site specific	• The site-specific areas vulnerable to flooding that contain safety related equipment are listed. Reference 6, Sections 3.1, 3.2, 3.3, 3.4, 3.5, 3.6, and 3.7 were used to develop the list of applicable areas. The areas listed in reference 6 are a summary of the analysis documented in reference 7, section 5.4.2 and reference 8, section 5.6.2.2.		
Difference	• Generic Difference Number 1 • Added "the following" to improve the grammar following insertion of the site-specific list.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
7	(Site-Specific) occurrences affecting the PROTECTED AREA.	HU1.7	High lake level: ultimate heat sink level GREATER THAN 590 ft elevation affecting the PROTECTED AREA. **OR** Low lake level: ultimate heat sink level LESS THAN OR EQUAL TO 572 ft elevation.
Site specific	• Added high lake level OR low lake level as specific occurrences that could affect the PROTECTED AREA. Actions for the high lake level are specified in reference 2, section 4.11. Reference 10, Section 5, steps 15-21 specifies the actions for low lake level.		
Difference	• Generic Difference Number 1, 7, 9		
Deviation	• None		

Hazards Justification Matrix

HU1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Added information on the National Earthquake Information Center in the basis for HU1.1.	1. Added contact information for the National Earthquake Information Center since onsite instrumentation is not immediately available to validate the seismic disturbance.
2. Added the basis of selecting 95 mph as the threshold for HU1.2.	2. Full-scale on the instrumentation used for classification is 100 mph. A value of 95 mph was chosen as the classification threshold; as this will still be on-scale.
3. Added a discussion on high and low lake water level to the basis for HU1.7	3. Information was added to clarify the limits for the site-specific occurrences. The limits on high and low lake level are consistent with Off Normal Procedures. The EAL limits were established to provide warning of the potential for external flooding and loss of the ultimate heat sink.
Difference	1, 2, 3 • The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HU2	FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection.	HU2	FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection.
Mode App.	All		All
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	FIRE in buildings or areas contiguous to any of the following (site-specific) areas not extinguished within 15 minutes of control room notification or verification of a control room alarm: (Site-specific) list	HU2.1	FIRE in buildings or areas contiguous to any of the following (Table H-1) areas not extinguished within 15 minutes of Control Room notification or verification of a Control Room alarm. Table H-1 Plant VITAL AREAS • Containment Structure • Auxiliary Building • Turbine Building • Screenhouse
Site specific	• Added Table H-1, Plant VITAL AREAS, as the list of applicable areas for this EAL. Table H-1 was developed based on reference 1, attachment 38. This attachment is derived from the Appendix R safe shutdown analysis and lists those areas considered vital for safe shutdown.		
Difference	• Generic Difference Number 1		
Deviation	• None		

Hazards Justification Matrix

HU2 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Added the word non-contiguous to the description of fire areas excluded from this EAL.	1. Some administrative buildings at PNP are contiguous to the Table H-1 areas. Fires in these buildings should be classified under this EAL. The basis addition clarifies this intent.
2. Changed the reference from HA4 to HA2 for escalation.	2. The referenced EAL number and title in NEI 99-01 are incorrect.

Difference	1, 2 • The additional information does not impact the classification or intent of the associated EAL(s).
Deviation	• None

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HU3	Release of Toxic or Flammable Gases Deemed Detrimental to Normal Operation of the Plant.	HU3	Release of Toxic or Flammable Gases Deemed Detrimental to Normal Operation of the Plant.
Mode App.	All		All
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	Report or detection of toxic or flammable gases that has or could enter the site area boundary in amounts that can affect NORMAL PLANT OPERATIONS.	HU3.1	Report or detection of toxic or flammable gases that has or could enter the site area boundary in amounts that can affect NORMAL PLANT OPERATIONS.
Site specific	• None		
Difference	• Generic Difference Number 1		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2	Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event.	HU3.2	Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event.
Site specific	• None		
Difference	• Generic Difference Number 1		
Deviation	• None		

Hazards Justification Matrix

HU3 – Basis Justification		
PNP Specific Additions/Deletions		Justification
• None		• None
Difference	• None	
Deviation	• None	

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HU4	Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant.	HU4	Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant.
Mode App.	All		All
Difference	• None		
Deviation	• None		

Hazards Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	Security events as determined from (site-specific) Safeguards Contingency Plan and reported by the (site-specific) security shift supervision	HU4.1	Security Shift Leader reports ANY of the following: • Suspected SABOTAGE device discovered within the plant PROTECTED AREA • Suspected SABOTAGE device discovered outside the PROTECTED AREA or in the plant switchyard • Confirmed tampering with safety-related equipment • A HOSTAGE/EXTORTION situation that disrupts NORMAL PLANT OPERATIONS • CIVIL DISTURBANCE or strike which disrupts NORMAL PLANT OPERATIONS • Internal disturbance that is <u>not</u> short lived or that is not a harmless outburst involving ANY individuals within the PROTECTED AREA • Malevolent use of a vehicle outside the PROTECTED AREA which disrupts NORMAL PLANT OPERATIONS
Site specific	• Added the list of site-specific security events. The list of events was developed by comparing the site-specific Safeguards Contingency Plan to the examples in the basis. The examples were then moved into the EAL to provide the decision maker with the appropriate list without the need to reference safeguards information. • Inserted the title of "Security Shift Leader" as the security shift supervision to report the events.		
Difference	• Generic Difference Number 1 • Restructured the EAL to allow listing the specific events.		
Deviation	• None		

Hazards Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2	A credible site specific security threat notification.	HU4.2	A credible site specific security threat notification.
Site specific	• None		
Difference	• Generic Difference Number 1		
Deviation	• None		

HU4 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Added the title "Security Shift Leader."	1. The site-specific title of the security shift supervision was added.
2. Added the title "Security Plan."	2. The site-specific title of the security plan was added.
3. Deleted the example information.	3. The site-specific events were added to EAL HU4.1
Difference	1, 2, 3 • The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HU5	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a NOUE.	HU5	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a UE.
Mode App.	All		All
Difference	• Generic Difference Number 4		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.	HU5.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.
Site specific	• None		
Difference	• Generic Difference Number 1		
Deviation	• None		

HU5 – Basis Justification		
PNP Specific Additions/Deletions		**Justification**
• None		• None
Difference	• None	
Deviations	• None	

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HA1	Natural and Destructive Phenomena Affecting the Plant VITAL AREA.	HA1	Natural and Destructive Phenomena Affecting the Plant VITAL AREA.
Mode App.	All		All
Difference	• None		
Deviation	• None		

Hazards Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	(Site-Specific) method indicates Seismic Event greater than Operating Basis Earthquake (OBE).	HA1.1	Seismic event GREATER THAN 0.1 g Operating Basis Earthquake (OBE) as indicated by VISIBLE DAMAGE to any of the following (Table H-2) plant structures / equipment. **Table H-2** **VITAL AREAS/Safe Shutdown Equipment** • Containment Structure • Auxiliary Building • Turbine Building • Screenhouse • Safety Injection Refueling Water Tank (T-58) • Condensate Storage Tank (T-2) • Primary System Makeup Tank (T-81) • Fuel Oil Storage Tank (T-10A)
Site specific	• The criterion of VISIBLE DAMAGE was added to distinguish a seismic event greater than OBE. • Table H-2 was inserted to identify the equipment required for safe shutdown that could be impacted with an OBE. Table H-2 was developed based on reference 11, attachment 38. This attachment is derived from the Appendix R safe shutdown analysis and lists those areas considered vital for safe shutdown. The Safety Injection Refueling Water Tank and Condensate Storage Tank were added as two safety related tanks located outside. Although these tanks are considered part of the Auxiliary Building and Turbine Building respectively, they are listed to ensure the decision maker evaluates them. The inclusion of these tanks is consistent with the list of components listed in NEI 99-01, HA1, EAL 2.		
Difference	• Generic Difference Number 1, 7 • The EAL was rearranged to include both the OBE value and the method of detection. • Inserted the value of 0.1g as the OBE to make the information readily available to the decision maker. Reference 1, section 2.4.4 provides the basis of the 0.1g OBE.		
Deviation	• None		

Hazards Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2	Tornado or high winds greater than (site-specific) mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures / equipment or Control Room indication of degraded performance of those systems. • Reactor Building • Intake Building • Ultimate Heat Sink • Refueling Water Storage Tank • Diesel Generator Building • Turbine Building • Condensate Storage Tank • Control Room • Other (Site-Specific) Structures.	HA1.2	Tornado or high winds GREATER THAN 95 mph within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following (Table H-2) plant structures / equipment or Control Room indication of degraded performance of those systems.
Site specific	• The site-specific value of "95 mph" wind speed was inserted to meet the FSAR 100 mph threshold within limitations of site instrumentation. Site instrumentation will only read up to 100 mph. A margin of 5 mph is considered an acceptable value for the instrumentation's accuracy. Reference 3, Section 5.3 "Wind And Tornado Loadings" lists the 100 mph design limit. • Table H-2 was used to replace the list of structures and components listed in the NEI 99-01 EAL. The structures and components listed in Table H-2 include all of those listed in the NEI EAL. For instance, the Auxiliary Building includes the Control Room and the Diesel Generator Building and the Screenhouse includes the Ultimate Heat Sink. Table H-2 was developed based on reference 11, attachment 38. This attachment is derived from the Appendix R safe shutdown analysis and lists those areas considered vital for safe shutdown. The Safety Injection Refueling Water Tank and Condensate Storage Tank were added as two safety related tanks located outside. Although these tanks are considered part of the Auxiliary Building and Turbine Building respectively, they are listed to ensure the decision maker evaluates them. The inclusion of these tanks is consistent with the components listed in NEI 99-01.		
Difference	• Generic Difference Number 1, 7		
Deviation	• None		

Hazards Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
3	Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following plant structures or equipment therein or control indication of degraded performance of those systems: • Reactor Building • Intake Building • Ultimate Heat Sink • Refueling Water Storage Tank • Diesel Generator Building • Turbine Building • Condensate Storage Tank • Control Room • Other (Site-Specific) Structures.	HA1.3	Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any of the following (Table H-2) plant structures or equipment therein or Control Room indication of degraded performance of those systems.
Site specific	• Table H-2 was used to replace the list of structures and components listed in the NEI 99-01 EAL. The structures and components listed in Table H-2 include all of those listed in the NEI EAL. For instance, the Auxiliary Building includes the Control Room and the Diesel Generator Building and the Screenhouse includes the Ultimate Heat Sink. Table H-2 was developed based on reference 11, attachment 38. This attachment is derived from the Appendix R safe shutdown analysis and lists those areas considered vital for safe shutdown. The Safety Injection Refueling Water Tank and Condensate Storage Tank were added as two safety related tanks located outside. Although these tanks are considered part of the Auxiliary Building and Turbine Building respectively, they are listed to ensure the decision maker evaluates them. The inclusion of these tanks is consistent with the components listed in NEI 99-01.		
Difference	• Generic Difference Number 1		
Deviation	• None		

Hazards Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
4	Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any of the following plant areas: (site-specific) list.	HA1.4	Turbine failure-generated missiles result in any VISIBLE DAMAGE to or penetration of any of the following (Table H-2) plant areas.
Site specific	• Table H-2 was developed based on reference 11, attachment 38. This attachment is derived from the Appendix R safe shutdown analysis and lists those areas considered vital for safe shutdown. The Safety Injection Refueling Water Tank and Condensate Storage Tank were added as two safety related tanks located outside. Although these tanks are considered part of the Auxiliary Building and Turbine Building respectively, they are listed to ensure the decision maker evaluates them. The inclusion of these tanks is consistent with the components listed in NEI 99-01, HA1, EAL1.		
Difference	• Generic Difference Number 1		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
5	Uncontrolled flooding in (site-specific) areas of the plant that results in degraded safety system performance as indicated in the control room or that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment.	HA1.5	Uncontrolled flooding in areas of the plant listed below that results in degraded safety system performance as indicated in the Control Room or that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment: • Emergency Diesel Generator Rooms • Engineered Safeguards Rooms • Auxiliary Feedwater Pump Room • Switchgear Room 1C • Screen House • Component Cooling Water Pump Room
Site specific	• The site-specific areas vulnerable to flooding that contain safety related equipment are listed. Reference 6, Sections 3.1, 3.2, 3.3, 3.4, 3.5, 3.6, and 3.7 were used to develop the list of applicable areas. The areas listed in reference 6 are a summary of the analysis documented in reference 7, section 5.4.2 and reference 8, section 5.6.2.2.		
Difference	• Generic Difference Number 1 • Added the words "listed below" to allow listing the areas at the end of the EAL.		
Deviation	• None		

Hazards Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
6	(Site-Specific) occurrences within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to plant structures containing equipment necessary for safe shutdown, or has caused damage as evidenced by control room indication of degraded performance of those systems.	HA1.6	The following occurrences within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to plant structures containing equipment necessary for safe shutdown, or has caused damage as evidenced by Control Room indication of degraded performance of those systems. High lake level: ultimate heat sink level GREATER THAN 594 ft elevation OR Low lake level: ultimate heat sink level LESS THAN OR EQUAL TO 569 ft elevation.
Site specific	• Added high lake level **OR** low lake level as specific occurrences that could affect the PROTECTED AREA. Reference 7, section 5.4.1.1 provides the basis of the high lake level limit. Actions for the high lake level are specified in reference 2, section 4.11. Reference 10, section 5, steps 15-21 specify the actions for low lake level.		
Difference	• Generic Difference Number 1 • "The following" was added to allow listing the site-specific occurrences at the end of the EAL.		
Deviation	• None		

Hazards Justification Matrix

HA1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Added information on the National Earthquake Information Center in the basis for HA1.1. Also added the value of the OBE (0.1g).	1. Added contact information for the National Earthquake Information Center since onsite instrumentation is not immediately available to validate the seismic disturbance. The magnitude of the OBE was added as site-specific information.
2. Added the basis of selecting 95 mph as the threshold for HU1.2.	2. Full-scale on the instrumentation used for classification is 100 mph. A value of 95 mph was chosen as the classification threshold, as this will still be on-scale.
3. Inserted reference to Table H-2.	3. Table H-2 was developed based on reference 11, attachment 38. This attachment is derived from the Appendix R safe shutdown analysis and lists those areas considered vital for safe shutdown. The Safety Injection Refueling Water Tank and Condensate Storage Tank were added as two safety related tanks located outside. Although these tanks are considered part of the Auxiliary Building and Turbine Building respectively, they are listed to ensure the decision maker evaluates them. The inclusion of these tanks is consistent with the list of components listed in NEI 99-01, HA1, EAL1.
4. Added a discussion on high and low lake water level to the basis for HA1.6	4. Information was added to clarify the limits for the site-specific occurrences. The limits on high and low lake level are consistent with Off Normal Procedures. The EAL limits were established to provide warning of the potential for external flooding and loss of the ultimate heat sink.
Difference	1, 2, 3, 4, 5 • The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HA2	FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown.	HA2	FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown.
Mode App.	All		All
Difference	• None		
Deviation	• None		

Hazards Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	FIRE or EXPLOSION in any of the following (site-specific) areas: (Site-specific) list **AND** Affected system parameter indications show degraded performance or plant personnel report VISIBLE DAMAGE to permanent structures or equipment within the specified area.	HA2.1	FIRE or EXPLOSION in any of the following (Table H-2) areas. **Table H-2** **VITAL AREAS/Safe Shutdown Equipment** • Containment Structure • Auxiliary Building • Turbine Building • Screenhouse • Safety Injection Refueling Water Tank (T-58) • Condensate Storage Tank (T-2) • Primary System Makeup Tank (T-81) • Fuel Oil Storage Tank (T-10A) **AND** Affected system parameter indications show degraded performance OR plant personnel report VISIBLE DAMAGE to permanent structures or equipment within the specified area.
Site specific	• Table H-2 was developed based on reference 1, attachment 38. This attachment is derived from the Appendix R safe shutdown analysis and lists those areas considered vital for safe shutdown. The Safety Injection Refueling Water Tank and Condensate Storage Tank were added as two safety related tanks located outside. Although these tanks are considered part of the Auxiliary Building and Turbine Building respectively, they are listed to ensure the decision maker evaluates them. The inclusion of these tanks is consistent with the list of components listed in NEI 99-01, HA1, EAL1.		
Difference	• Generic Difference Number 1		
Deviation	• None		

Hazards Justification Matrix

HA2 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Inserted reference to Table H-2.	1. Table H-2 was developed based on reference 10, attachment 38. This attachment is derived from the Appendix R safe shutdown analysis and lists those areas considered vital for safe shutdown. The Safety Injection Refueling Water Tank and Condensate Storage Tank were added as two safety related tanks located outside. Although these tanks are considered part of the Auxiliary Building and Turbine Building respectively, they are listed to ensure the decision maker evaluates them. The inclusion of these tanks is consistent with the list of components listed in NEI 99-01, HA1, EAL1.
Difference	1. • The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HA3	Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown.	HA3	Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown.
Mode App.	All		All
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	Report or detection of toxic gases within or contiguous to a VITAL AREA in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).	HA3.1	Report or detection of toxic gases within or contiguous to a VITAL AREA (Table H-1) in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH). Table H-1 Plant VITAL AREAS • Containment Structure • Auxiliary Building • Turbine Building • Screenhouse
Site specific	• None		
Difference	• Generic Difference Number 1 • Added Table H-1, Plant VITAL AREAS, as the list of applicable areas for this EAL. Table H-1 was developed based on reference 1, attachment 38. This attachment is derived from the Appendix R safe shutdown analysis and lists those areas considered vital for safe shutdown.		
Deviation	• None		

Hazards Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2	Report or detection of gases in concentration greater than the LOWER FLAMMABILITY LIMIT within or contiguous to a VITAL AREA.	HA3.2	Report or detection of gases in concentration greater than the LOWER FLAMMABILITY LIMIT within or contiguous to a VITAL AREA (Table H-1).
Site specific	• None		
Difference	• Generic Difference Number 1 • Added Table H-1, Plant VITAL AREAS, as the list of applicable areas for this EAL. Table H-1 was developed based on reference 1, attachment 38. This attachment is derived from the Appendix R safe shutdown analysis and lists those areas considered vital for safe shutdown.		
Deviation	• None		

HA3 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Removed the service water pump example.	1. The appropriate site-specific area was included in Table H-1.
Difference	1. • The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HA4	Confirmed Security Event in a Plant PROTECTED AREA.	HA4	Confirmed Security Event in a Plant PROTECTED AREA.
Mode App.	All		All
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	INTRUSION into the plant PROTECTED AREA by a HOSTILE FORCE.	HA4.1	INTRUSION into the plant PROTECTED AREA by a HOSTILE FORCE.
Site specific	• None		
Difference	• Generic Difference Number 1		
Deviation	• None		

Hazards Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording

2	Other security events as determined from (site-specific) Safeguards Contingency Plan and reported by the (site-specific) security shift supervision	HA4.2	Security Shift Leader reports any of the following:
			• SABOTAGE device discovered in the plant PROTECTED AREA
			• Standoff attack on the PROTECTED AREA by a HOSTILE FORCE (i.e., Sniper)
			• ANY Security event of increasing severity that persists for GREATER THAN 30 min.:
			  • Credible BOMB threats
			  • HOSTAGE/EXTORTION
			  • Suspicious FIRE OR EXPLOSION
			  • Significant Security System Hardware Failure
			  • Loss of Guard Post Contact

Site specific	• Added the list of site-specific security events. The list of events was developed by comparing the site-specific Security Plan to the examples in the basis. The examples were then moved into the EAL to provide the decision maker with the appropriate list without the need to reference safeguards information. • Inserted the title of "Security Shift Leader" as the security shift supervision to report the events.		
Difference	• Generic Difference 1. • Restructured the EAL to allow listing the specific events.		
Deviation	• None		

HA4 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Generic Difference 4.	1. See Generic Difference 4 justification.
2. Added the Security Plan as the source of the listed events.	2. The site-specific title of the security plan was added.
3. Deleted the example information.	3. The site-specific events were added to EAL HA4.2
4. Added the title "Security Shift Leader."	4. The site-specific title of the security shift supervision was added.
Difference	1, 2, 3, 4 • The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HA5	Control Room Evacuation Has Been Initiated.	HA5	Control Room Evacuation Has Been Initiated.
Mode App.	All		All
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	Entry into (site-specific) procedure for control room evacuation.	HA5.1	Entry into ONP-25.2, "Alternate Safe Shutdown" for Control Room evacuation.
Site specific	• Listed ONP-25.2, "Alternate Safe Shutdown" as the site-specific procedure for Control Room evacuation.		
Difference	• Generic Difference Number 1		
Deviation	• None		

HA5 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Added the site-specific procedure ONP-25.2.	1. ONP-25.2 is the site-specific procedure that provides instructions for evacuating the Control Room and establishing plant control at the auxiliary hot shutdown panels.
Difference	1. • The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HA6	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert.	HA6	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert.
Mode App.	All		All
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels .	HA6.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels .
Site specific	• None		
Difference	• Generic Difference 1.		
Deviation	• None		

HA6 – Basis Justification		
PNP Specific Additions/Deletions		**Justification**
• None		
Difference	• None	
Deviations	• None	

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HS1	Confirmed Security Event in a Plant VITAL AREA.	HS1	Confirmed Security Event in a Plant VITAL AREA.
Mode App.	All		All
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	INTRUSION into the plant VITAL AREA by a HOSTILE FORCE.	HS1.1	INTRUSION into the plant VITAL AREA by a HOSTILE FORCE.
Site specific	• None		
Difference	• Generic Difference Number 1		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2	Other security events as determined from (site-specific) Safeguards Contingency Plan and reported by the (site-specific) security shift supervision	HS1.2	Security Shift Leader reports any of the following: • SABOTAGE discovered in a VITAL AREA. • HOSTAGE/EXTORTION in a VITAL AREA.
Site specific	• Added the list of site-specific security events. The list of events was developed by comparing the site-specific Security Plan to the examples in the basis. The examples were then moved into the EAL to provide the decision maker with the appropriate list without the need to reference safeguards information. • Inserted the title of "Security Shift Leader" as the security shift supervision to report the events.		
Difference	• Generic Difference Number 1 • The EAL was reworded for clarity.		
Deviation	• None		

Hazards Justification Matrix

HS1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Deleted the example information.	1. The site-specific events were added to EAL HS1.2
2. Added the title "Security Shift Leader."	2. The site-specific title of the security shift supervision was added.
Difference	1, 2 • The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HS2	Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established.	HS2	Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established.
Mode - App.	All		All
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	Control room evacuation has been initiated. **AND** Control of the plant cannot be established per (site-specific) procedure within (site-specific) minutes.	HS2.1	Control Room evacuation has been initiated. **AND** Control of the plant cannot be established per ONP-25.2, Alternate Safe Shutdown within 15 minutes.
Site specific	• Added "ONP-25.2, Alternate Safe Shutdown" as the site-specific procedure for Control Room evacuation. • Added "15 minutes" as the site-specific limit for establishing control per ONP-25.2. ONP-25.2, section 6, specifies times for completing manual operator actions to control the plant. All of the times for these actions are greater than 15 minutes with the exception of isolating atmospheric steam dumps. Steam dumps need to be isolated within six minutes to avoid overcooling the PCS but this action would be completed prior to establishing control at C-150/C-150A. The 15 minute limit is established to ensure control is established at C-150/C-150A in sufficient time to allow completion of the remaining actions in ONP-25.2.		
Difference	• Generic Difference Number 1		
Deviation	• None		

Hazards Justification Matrix

HS2 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Deleted basis information on determining time for reestablishing control.	1. Example information was replaced with the site-specific information.
2. Added the site-specific procedure ONP-25.2. and the basis for the 15 minute limit.	2. ONP-25.2 is the site-specific procedure that provides instructions for evacuating the Control Room and establishing plant control at the auxiliary hot shutdown panels. The 15 minute limit is established to ensure control is established at the Auxiliary Hot Shutdown panel in sufficient time to allow completion of the actions listed in ONP-25.2.
3. Removed BWR information.	3. Retained the PWR information and removed the BWR information for ease of use.
Difference	1, 2, 3 • The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HS3	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency.	HS3	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency.
Mode App.	All		All
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	HS3.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.
Site specific	• None		
Difference	• Generic Difference Number 1		
Deviation	• None		

HS3 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
• None	
Difference	• None
Deviations	• None

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HG1	Security Event Resulting in Loss Of Physical Control of the Facility.	HG1	Security Event Resulting in Loss Of Physical Control of the Facility.
Mode App.	All		All
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	A HOSTILE FORCE has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions.	HG1.1	A HOSTILE FORCE has taken control of plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions as indicated by loss of physical control of EITHER: A VITAL AREA such that operation of equipment required for safe shutdown is lost **OR** Spent fuel pool cooling systems if imminent fuel damage is likely (e.g., freshly off-loaded reactor core in the pool).
Site specific	• None		
Difference	• Generic Difference Number 1 • Added "as indicated by loss of physical control of EITHER: a VITAL AREA such that operation of equipment required for safe shutdown is lost OR Spent fuel pool cooling systems if imminent fuel damage is likely (e.g., freshly off-loaded reactor core in the pool)" as a site specific list of required equipment. This information was pulled from the basis into the EAL wording to facilitate consistent and accurate classification without the need to reference the basis information.		
Deviation	• None		

Hazards Justification Matrix

HG1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. Removed BWR information.	1. Retained the PWR information and removed the BWR information for ease of use.
2. Deleted the example information.	2. The site-specific events were added to EAL HG1.1
Difference	1, 2 • The additional information does not impact the classification or intent of the associated EAL(s).
Deviations	• None

Hazards Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
HG2	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency.	HG2	Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency.
Mode App.	All		All
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	HG2.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.
Site specific	• None		
Difference	• Generic Difference Number 1		
Deviation	• None		

HG2 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
• None	
Difference	• None
Deviations	• None

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
SU1	Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes	SU1	Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Startup, Hot Standby, Hot Shutdown
Difference	• Generic Difference Number 7		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	Loss of power to (site-specific) transformers for greater than 15 minutes. **AND** At least (site-specific) emergency generators are supplying power to emergency busses.	SU1.1.	Loss of all offsite power to both Vital 2400 VAC busses 1C and 1D for GREATER THAN 15 minutes. **AND** Both emergency diesel generators are supplying power to both Vital 2400 VAC busses 1C and 1D.
Site specific	• "Vital 2400 VAC busses 1C and 1D" are the site specific essential busses • "Both" is the site-specific number of generators needed to power "both" emergency busses.		
Difference	• Generic Difference Number 1, 7 • "each Vital 2400 VAC bus" added for user clarity. • The NEI example EAL condition "Loss of power to (site-specific) transformers for greater than 15 minutes" has been changed to "Loss of all offsite power to Vital 2400 VAC busses 1C and 1D for GREATER THAN 15 min." Adding the words "all offsite" to the EAL focuses the classification on the loss of offsite power to the essential busses. Additionally, the words "all offsite" brings information from the basis into the EAL and align the EAL directly to the IC. The PNP wording focuses the classification on the loss of offsite power capability rather than the status of one or more transformers that may or may not be capable of powering the essential buses (e.g. in the event of a fault condition between the transformer and the bus). This simplifies the EAL wording and concisely meets the intent of the NEI IC SU1. The combination of changing transformers to busses, and adding the "all offsite" to the EAL was done to provide the user all the information necessary within the EAL to accurately classify the event.		
Deviation	• None		

System Malfunctions Justification Matrix

SU1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. The PNP specific basis includes additional information related to the power supplies for the Vital 2400 VAC busses.	1. The additional basis information supports the determination of a loss of offsite power specified in the EAL.
2. Deleted reference to companion unit AC cross-tie capability.	2. PNP is a single unit plant.

Difference	1. Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EALs.
	2. Deleted reference to companion unit AC cross-tie capability as PNP is a single unit plant.
Deviations	• None

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
SU2	Inability to Reach Required Shutdown Within Technical Specification Limits	SU2	Inability to Reach Required Shutdown Within Technical Specification Limits
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Startup, Hot Standby, Hot Shutdown
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	Plant is not brought to required operating mode within (site-specific) Technical Specifications LCO Action Statement Time	SU2.1.	Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time
Site specific	• Generic Difference Number 1 • The site-specific Technical Specification LCO Action Statement Time has been left blank to imply the inability to reach the required mode stated in <u>any</u> of the various completion times warrants event classification.		
Difference	• None		
Deviation	• None		

SU2 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
• Minor grammatical revisions.	• Enhance user interpretation.
Difference	• Minor revisions to enhance basis readability.
Deviations	• None

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
SU3	UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for Greater Than 15 Minutes	SU3	UNPLANNED Loss of Most or All Safety System Annunciation or Indication in the Control Room for GREATER THAN 15 Minutes
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Startup, Hot Standby, Hot Shutdown
Difference	• Generic Difference Number 7		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	UNPLANNED loss of most or all (site-specific) annunciators or indicators associated with safety systems for greater than 15 minutes	SU3.1	UNPLANNED loss of most or all annunciators or indicators associated with safety systems for GREATER THAN 15 minutes on the following: • EC-02/12 – Reactor/PCPs • EC-03/13 – Safety Injection/Containment Cooling & Isolation systems / Ventilation • EC-04 – Electrical • EC-06 – RPS • EC-08 – Service Water/Component Cooling • EC-11 (Rear) – Rad Monitors • EC-11A (Front) – Control Room HVAC / Reactor Vessel Level / Core Exit Thermocouples • EC-27 - Thermal Margin Monitors
Site specific	• The site-specific list of Electrical Cabinets (EC) houses the PNP safety system annunciators and indicators.		
Difference	• Generic Difference Number 1, 7 • Added "on the following:" for readability.		
Deviation	• None		

System Malfunctions Justification Matrix

SU3 – Basis Justification		
PNP Specific Additions/Deletions		**Justification**
• Minor grammatical revisions.		• Enhance user interpretation.
Difference	• Minor revisions to enhance basis readability.	
Deviations	• None	

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
SU4	Fuel Clad Degradation.	SU4	Fuel Clad Degradation.
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Startup, Hot Standby, Hot Shutdown
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	(Site-specific) radiation monitor readings indicating fuel clad degradation greater than Technical Specification allowable limits.	SU4.1.	Any of the following radiation monitors with a VALID PPC urgent alarm indicating fuel clad degradation GREATER THAN Technical Specification allowable limits. Containment Isolation High Radiation Monitor - RIA-1805 Containment Isolation High Radiation Monitor - RIA-1806 Containment Isolation High Radiation Monitor - RIA-1807 Containment Isolation High Radiation Monitor - RIA-1808
Site specific	• The Containment Isolation High Radiation Monitors in high alarm are the "site-specific" fuel clad degradation indications.		
Difference	• Generic Difference Number 1, 7 • PNP uses "with a VALID PPC urgent alarm" as the radiation monitor "reading" due to: 1) having these alarm setpoints at a value that indicates fuel clad degradation, 2) the need to provide the operators with an active prompt versus a passive reading from an instrument that is not readily accessed, and 3) to align this syntax with that used for other radiation monitor readings in the Abnormal Radiation EALs.		
Deviation	• For the Containment Isolation High Radiation monitors a special test was conducted that verified that with approximately 1% failed fuel the Containment Isolation High Radiation monitors would indicate 4.5 Rem/hour. The PPC urgent alarm is set at one tenth of that value, 0.45 Rem/hour, indicating fuel clad degradation. This deviation is appropriate for PNP as it provides equivalent protection of the public health and safety in the absence of an installed fuel element failure radiation monitor. Additionally, the setpoint selected is high enough to be discriminatory yet low enough to allow escalation to an Alert in the Fission Product Barrier Degradation category. Therefore use of these radiation monitors does not decrease the effectiveness of the EAL scheme.		

System Malfunctions Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2.	(Site-specific) coolant sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits.	SU4.2.	Coolant sample activity GREATER THAN OR EQUAL TO 40 µCi/gm dose equivalent I-131 indicating fuel clad degradation
Site specific	• "GREATER THAN 40 µCi/gm dose equivalent I-131" is the site-specific Technical Specification allowable limit.		
Difference	• Generic Difference Number 1, 7 • "Technical Specification allowable limits" was deleted as it duplicates the site specific Technical Specification value already listed in the EAL.		
Deviation	• None		

System Malfunctions Justification Matrix

SU4 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. The PNP specific basis includes additional information related to the radiation monitors' alarms that would prompt operations to request a PCS sample and activity analysis.	1. PNP has no radiation monitor that could definitively indicate fuel-clad degradation. Therefore, a prompt has been provided to confirm or deny fuel clad degradation.
2. The PNP specific basis includes additional information related to the appropriateness of the coolant sample Technical Specification as a limit for this EAL.	2. The additional basis information supports the determination of fuel cladding degradation specified in the EAL.

Difference	1. Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EAL.
	2. Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EAL.
Deviations	• None

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
SU5	RCS Leakage.	SU5	PCS Leakage.
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Startup, Hot Standby, Hot Shutdown
Difference	• Generic Difference Number 16		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	Unidentified or pressure boundary leakage greater than 10 gpm.	SU5.1	Unidentified or pressure boundary leakage GREATER THAN 10 gpm.
Site specific	• None		
Difference	• Generic Difference Number 1, 7		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2	Identified leakage greater than 25 gpm	SU5.2	Identified leakage GREATER THAN 25 gpm
Site specific	• None		
Difference	• Generic Difference Number 1, 7		
Deviation	• None		

System Malfunctions Justification Matrix

SU5 – Basis Justification		
PNP Specific Additions/Deletions		Justification
• Minor grammatical revisions.		• Enhance user interpretation.
Difference	• Minor revisions to enhance basis readability.	
Deviations	• None	

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
SU6	UNPLANNED Loss of All Onsite or Offsite Communications Capabilities	SU6	UNPLANNED Loss of All Onsite or Offsite Communications Capabilities
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Startup, Hot Standby, Hot Shutdown
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	Loss of all (site-specific list) onsite communications capability affecting the ability to perform routine operations	SU6.1.	Loss of all Table C-1 onsite communications capability affecting the ability to perform routine operations **Table C-1 Onsite Communications Systems** • Telephone system • Onsite/offsite radio system • Public address system
Site specific	• Table C-1 is the site-specific Onsite Communications Systems list		
Difference	• Generic Difference Number 1		
Deviation	• None		

System Malfunctions Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2.	Loss of all (site-specific list) offsite communications capability.	SU6.2.	Loss of all Table C-2 offsite communications capability. **Table C-2 Onsite Communications Systems** • Telephone system • Power failure phones • FTS phone system • Satellite phone
Site specific	• Table C-2 is the site-specific Onsite Communications Systems list •		
Difference	• Generic Difference Number 1		
Deviation	• None		

SU6 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
• The PNP specific basis includes the site-specific Table C-1 and C-2 references related to a loss of communications.	• The additional basis information supports the determination of a loss of communications specified in the EALs.
Difference	• Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EALs.
Deviations	• None

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
SU8	Inadvertent Criticality.	SU8	Inadvertent Criticality.
Mode App.	Hot Standby, Hot Shutdown		Hot Standby, Hot Shutdown
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	An UNPLANNED extended positive period observed on nuclear instrumentation.		(Not used)
Site specific	• None		
Difference	• PNP is a PWR and has no "period" nuclear instrumentation.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2.	An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.	SU8.1	An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.
Site specific	• None		
Difference	• Generic Difference Number 1		
Deviation	• None		

System Malfunctions Justification Matrix

SU8 – Basis Justification		
PNP Specific Additions/Deletions		Justification
• Deleted reference to period monitors.		• PNP is a PWR and has no period nuclear instrumentation.
Difference	• BWR-specific EAL not used.	
Deviations	• None	

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
SA2	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 Successful	SA2	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 Successful
Mode App.	Power Operation, Startup, Hot Standby		Power Operation, Startup, Hot Standby
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	Indication(s) exist that indicate that reactor protection system setpoint was exceeded and automatic scram did not occur, and a successful manual scram occurred	SA2.1.	Indication(s) exist that a Reactor Protection System setpoint was exceeded **AND** RPS automatic trip did NOT occur **AND** A successful manual trip occurred from: • EC-02 Reactor Trip pushbutton OR • EC-06 Reactor Trip pushbutton OR • Reactor tripped from an ATWS trip signal
Site specific	• None		
Difference	• Generic Difference Number 1, 10, 11, 12, 15 • Capitalized "Reactor Protection System" for user clarity. • Inserted "RPS" for user clarity. • Replaced "scram" with "trip" to be consistent with site terminology. • Defined "a successful manual scram" to be from either of the readily accessible manual reactor trip pushbuttons, or the diverse ATWS signal.		
Deviation	• None		

System Malfunctions Justification Matrix

SA2 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. The PNP specific basis replaces the NEI discussion defining site-specific trip methodology and references related to what constitutes a successful reactor trip.	1. The additional basis information supports the determination of an RPS automatic trip failure with successful manual trip specified in the EALs.
2. The ATWS trip is included as a trip equivalent to the manual reactor trips.	2. The ATWS trip circuitry uses a diverse trip methodology. This trip methodology accomplishes a reactor trip via the same circuitry as the EC-06 Reactor Trip manual pushbutton.

Difference	1. Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EALs.
	2. Additional information was added to support ATWS as a diverse trip equivalent to the manual reactor trip pushbuttons.
Deviations	• None

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
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.	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.
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Startup, Hot Standby, Hot Shutdown
Difference	• None		
Deviation	• None		

System Malfunctions Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	UNPLANNED loss of most or all (site-specific) annunciators or indicators associated with safety systems for greater than 15 minutes. **AND** Either of the following: (a or b) a. A SIGNIFICANT TRANSIENT is in progress. **OR** b. Compensatory non-alarming indications are unavailable.	SA4.1.	UNPLANNED loss of most or all annunciators or indicators associated with safety systems for GREATER THAN 15 minutes on the following: • EC-02/12 – Reactor/PCPs • EC-03/13 – Safety Injection/Containment Cooling & Isolation systems / Ventilation • EC-04 – Electrical • EC-06 – RPS • EC-08 – Service Water/Component Cooling • EC-11 (Rear) – Rad Monitors • EC-11A (Front) – Control Room HVAC / Reactor Vessel Level / Core Exit Thermocouples • EC-27- Thermal Margin Monitors **AND** Either of the following: (a or b) a. A SIGNIFICANT TRANSIENT is in progress. OR b. Compensatory non-alarming indications are unavailable.
Site specific	• The site-specific list of Electrical Cabinets (EC) houses the PNP safety system annunciators and indicators.		
Difference	• Generic Difference Number 1, 7 • Added "on the following:" for readability.		
Deviation	• None		

System Malfunctions Justification Matrix

SA4 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
• The PNP specific basis includes the site-specific definition for what constitutes a significant transient and compensatory non-alarming indications.	• The additional basis information supports the determination of a loss of annunciation and indication with a transient in-progress and no compensatory indications specified in the EALs.

Difference	• Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EALs.
Deviations	• None

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
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	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
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Startup, Hot Standby, Hot Shutdown
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	AC power capability to site-specific essential busses reduced to a single power source for greater than 15 minutes **AND** Any additional single failure will result in station blackout.	SA5.1.	AC power capability to Vital 2400 VAC busses 1C and 1D reduced to only one of the following sources for GREATER THAN 15 minutes • Safeguard Transformer 1-1 • Start Up Transformer 1-2 • Station Power Transformer 1-2 • 1-1 Emergency Diesel Generator • 1-2 Emergency Diesel Generator **AND** Any additional single failure will result in station blackout.
Site specific	• "Vital 2400 VAC busses 1C and 1D" are the site-specific essential busses.		
Difference	• Generic Difference Number 1, 7 • Defined "a single power source" to be "only one of the following sources" and listed those sources for user clarity.		
Deviation	• None		

System Malfunctions Justification Matrix

SA5 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
1. The PNP specific basis includes additional information related to the power supplies for the Vital 2400 VAC busses.	1. The additional basis information supports the determination of a loss of offsite power specified in the EAL.
2. Deleted reference to companion unit AC cross-tie capability.	2. PNP is a single unit plant.

Difference	1. Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EALs.
	2. Deleted reference to companion unit AC cross-tie capability as PNP is a single unit plant.
Deviations	• None

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
SS1	Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses	SS1	Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Startup, Hot Standby, Hot Shutdown
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	Loss of power to (site-specific) transformers. **AND** Failure of (site-specific) emergency generators to supply power to emergency busses. **AND** Failure to restore power to at least one emergency bus within (site-specific) minutes from the time of loss of both offsite and onsite AC power	SS1.1.	Loss of all offsite power to Vital 2400 VAC buses 1C and 1D. **AND** Failure of both emergency diesel generators to supply power to Vital 2400 VAC busses. **AND** Failure to restore power to at least one Vital 2400 VAC bus within 15 minutes from the time of loss of both offsite and onsite AC power.
Site specific	• "Vital 2400 VAC buses 1C and 1D" are the site-specific essential busses • "both" is site-specific for the emergency "diesel" generators		
Difference	• Generic Difference Number 1 • Replaced "emergency" with "Vital 2400 VAC" for user clarity. • The NEI example EAL condition "Loss of power to (site-specific) transformers" has been changed to "Loss of all offsite power to Vital 2400 VAC buses 1C and 1D." Adding the words "all offsite" to the EAL focuses the classification on the loss of offsite power to the essential busses. Additionally, the words "all offsite" brings information from the basis into the EAL and align the EAL directly to the IC. The PNP wording focuses the classification on the loss of offsite power capability rather than the status of one or more transformers that may or may not be capable of powering the essential buses (e.g. in the event of a fault condition between the transformer and the bus). This simplifies the EAL wording and concisely meets the intent of the NEI IC SS1. The combination of changing transformers to busses, and adding the "all offsite" to the EAL was done to provide the user all the information necessary within the EAL to accurately classify the event.		
Deviation	• None		

System Malfunctions Justification Matrix

SS1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
• The PNP specific basis includes additional information related to the power supplies for the Vital 2400 VAC busses.	• The additional basis information supports the determination of a loss of offsite power specified in the EAL.

Difference	• Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EAL.
Deviations	• None

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
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	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
Mode App.	Power Operation, Startup		Power Operation, Startup
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	Indication(s) exist that automatic and manual scram were not successful.	SS2.1.	Indication(s) exist that automatic and manual trip were NOT successful.
Site specific	• None		
Difference	• Generic Difference Number 1, 11, 15		
Deviation	• None		

SS2 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
• The PNP specific basis includes the site-specific trip methodology and references related to what constitutes a successful and unsuccessful reactor trip.	• The additional basis information supports the determination of an RPS automatic trip failure with an unsuccessful manual trip specified in the EALs.
Difference	• Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EALs.
Deviations	• None

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
SS3	Loss of All Vital DC Power	SS3	Loss of All Vital DC Power
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Startup, Hot Standby, Hot Shutdown
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	Loss of All Vital DC Power based on (site-specific) bus voltage indications for greater than 15 minutes.	SS3.1.	Loss of all vital DC power based on LESS THAN 105 VDC on DC busses No. 1, ED-10 and No.2, ED-20 for GREATER THAN 15 minutes.
Site specific	• "LESS THAN 105 VDC on DC buses No. 1, ED-10 and No.2, ED-20" is the site-specific bus voltage indication for a loss of all Vital DC.		
Difference	• Generic Difference Number 1, 7		
Deviation	• None		

SS3 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
• The PNP specific basis includes additional information related to Station Battery capabilities.	• The additional basis information supports the determination of the loss of DC power specified in the EAL.
Difference	• Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EAL.
Deviations	• None

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
SS4	Complete Loss of Heat Removal Capability	SS4	Complete Loss of Heat Removal Capability
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Startup, Hot Standby, Hot Shutdown
Difference	• None		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	Loss of core cooling and heat sink (PWR).	SS4.1.	Loss of core cooling and heat sink.
Site specific	• None		
Difference	• Generic Difference Number 1 • Deleted "(PWR)" as it is redundant to PNP design.		
Deviation	• None		

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
2.	Heat Capacity Temperature Limit Curve exceeded (BWR).		(Not used)
Site specific	• PNP is a PWR and has no "Heat Capacity Temperature Limit Curve".		
Difference	• Not used		
Deviation	• None		

System Malfunctions Justification Matrix

SS4 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
• The PNP specific basis replaces the BWR-specific information with the PNP functions that must be lost.	• The additional basis information supports the determination of the loss of core cooling and heat sink specified in the EAL.
Difference	• Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EAL.
Deviations	• None

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
SS6	Inability to Monitor a SIGNIFICANT TRANSIENT in Progress.	SS6	Inability to Monitor a SIGNIFICANT TRANSIENT in Progress.
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Startup, Hot Standby, Hot Shutdown
Difference	• None		
Deviation	• None		

System Malfunctions Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	a. Loss of most or all (site-specific) annunciators associated with safety systems. **AND** b. Compensatory non-alarming indications are unavailable. **AND** c. Indications needed to monitor (site-specific) safety functions are unavailable. **AND** d. SIGNIFICANT TRANSIENT in progress.	SS6.1.	Loss of most or all annunciators associated with safety systems on the following: • EC 02/12 – Reactor/PCPs • EC-03/13 – Safety Injection/Containment Cooling & Isolation systems / Ventilation • EC-04 – Electrical • EC-06 – RPS • EC-08 – Service Water/Component Cooling • EC-11 (Rear) – Rad Monitors • EC-11A (Front) – Control Room HVAC / Reactor Vessel Level / Core Exit Thermocouples • EC-27- Thermal Margin Monitors **AND** SIGNIFICANT TRANSIENT in progress. **AND** Compensatory non-alarming indications are unavailable. **AND** Indications needed to monitor the ability to shut down the reactor, maintain the core cooled, maintain the primary coolant system intact, and maintain containment intact are unavailable.
Site specific	• The site-specific list of Electrical Cabinets (EC) houses the PNP safety system annunciators and indicators.. • "the ability to shut down the reactor, maintain the core cooled, maintain the primary coolant system intact, and maintain containment intact" is the site-specific list of safety functions.		
Difference	• Generic Difference Number 1 • "SIGNIFICANT TRANSIENT" placed 2nd on list to provide user with clear escalation path.		
Deviation	• None		

System Malfunctions Justification Matrix

SS6 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
• The PNP specific basis includes the site-specific definition for what constitutes a significant transient and compensatory non-alarming indications.	• The additional basis information supports the determination of an inability to monitor in-progress significant transients specified in the EALs.
Difference	• Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EALs.
Deviations	• None

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
SG1	Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses.	SG1	Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses.
Mode App.	Power Operation, Startup, Hot Standby, Hot Shutdown		Power Operation, Startup, Hot Standby, Hot Shutdown
Difference	• None		
Deviation	• None		

System Malfunctions Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	Loss of power to (site-specific) transformers. **AND** Failure of (site-specific) emergency diesel generators to supply power to emergency busses. **AND** Either of the following: (a or b) a. Restoration of at least one emergency bus within (site-specific) hours is <u>not</u> likely **OR** b. (Site-Specific) Indication of continuing degradation of core cooling based on Fission Product Barrier monitoring.	SG1.1.	Loss of all offsite power to Vital 2400 VAC busses 1C and 1D. **AND** Failure of both emergency diesel generators to supply power to Vital 2400 VAC busses. **AND** Either of the following: (a or b) a. Restoration of at least one Vital 2400 VAC bus within 4 hours is NOT likely OR b. Continuing degradation of core cooling based on Fission Product Barrier monitoring as indicated by Average of qualified CETs GREATER THAN 700 degrees F.
Site specific	• "Vital 2400 VAC buses 1C and 1D" are the site-specific essential busses • "both" is site-specific for the emergency diesel generators • "4 hours" is the site-specific value for the batteries design life during a loss of all AC. • "CETs GREATER THAN 700 degrees F" is the site-specific indication for continuing core cooling degradation.		
Difference	• Generic Difference Number 1, 15 • Replaced "emergency" with "Vital 2400 VAC" for user clarity. • The NEI example EAL condition "Loss of power to (site-specific) transformers" has been changed to "Loss of all offsite power to Vital 2400 VAC buses 1C and 1D." Adding the words "all offsite" to the EAL focuses the classification on the loss of offsite power to the essential busses. Additionally, the words "all offsite" brings information from the basis into the EAL and align the EAL directly to the IC. The PNP wording focuses the classification on the loss of offsite power capability rather than the status of one or more transformers that may or may not be capable of powering the essential buses (e.g. in the event of a fault condition between the transformer and the bus). This simplifies the EAL wording and concisely meets the intent of the NEI IC SG1. The combination of changing transformers to busses, and adding the "all offsite" to the EAL was done to provide the user all the information necessary within the EAL to accurately classify the event.		
Deviation	• None		

System Malfunctions Justification Matrix

SG1 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
• The PNP specific basis includes additional information related to the power supplies for the Vital 2400 VAC busses and use of the CETs for indicating degraded core cooling.	• The additional basis information supports the determination of a prolonged loss of offsite power specified in the EAL
Difference	• Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EAL.
Deviations	• None

System Malfunctions Justification Matrix

NEI IC#	NEI IC Wording	PNP IC#(s)	PNP IC Wording
SG2	Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core	SG2	Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core
Mode App.	Power Operation, Startup		Power Operation, Startup
Difference	• None		
Deviation	• None		

System Malfunctions Justification Matrix

NEI EAL#	NEI EAL Wording	PNP EAL#(s)	PNP EAL Wording
1.	Indications exist that automatic and manual scram were not successful. **AND** Either of the following: (a or b) a. Indication(s) exists that the core cooling is extremely challenged. **OR** b. Indication(s) exists that heat removal is extremely challenged.	SG2.1.	Indications exist that automatic and manual trip were NOT successful. **AND** Either of the following: (a or b) a. Average of qualified CETs GREATER THAN 1200 degrees F. **OR** b. Core and PCS Heat Removal safety function status acceptance criteria cannot be met.
Site specific	• None		
Difference	• Generic Difference Number 1, 11, 15 • "Average of qualified CETs GREATER THAN 1200 degrees F" is the site-specific indication that "core cooling is extremely challenged". • "Core and PCS Heat Removal safety function status acceptance criteria cannot be met" is the site-specific indication that "heat removal is extremely challenged".		
Deviation	• None		

System Malfunctions Justification Matrix

SG2 – Basis Justification	
PNP Specific Additions/Deletions	**Justification**
• The PNP specific basis includes additional information related to automatic and manual reactor trip failures, and extreme challenges to core cooling and heat removal.	• The additional basis information supports the determination of reactor trip failure with extreme core challenges specified in the EAL.

Difference	• Additional information was added to support EAL determination. The additional information does not impact the classification or intent of the associated EAL.
Deviations	• None

Justification Matrix Addendum #1
Justification for RVLMS Levels

The deviations applied to the EALs related to loss of PCS inventory accomplish the following:

- Apply the methodology used in the NEI 99-01, Revision 4 guidance.
- Utilize available plant specific indications to provide the end-user with indicated reactor vessel levels in addition to backup indications based on source range indication and containment radiation levels.

The following comparison of the NEI to PNP EALs for a loss of reactor vessel inventory should be used as a reference for discussions of the deviations in EALs CA1.1, CA2.1, CS1.1, CS1.2, CS2.1, CS2.2, CG1.2, Fission Product Barrier Matrix; Fuel Clad Potential Loss #4 and Containment Potential Loss #3.

NEI EAL	PNP Elevation	PNP EAL	PNP Available Indications
			628' 6" on LT-0105 @ 100%
			619' 6" on LT-0106 @ 100% 619' 2" on RVLMS Sensor #6
Alert bottom ID of the RCS loop	616' 5.5"	**Alert** 617'	617' on LT-0105/0106 @ 0%
SAE 6" below the bottom ID of the RCS loop	615' 11.5"	**SAE** 616' 6"	616' 6" on RVLMS Sensor #7
GE less than TOAF for > 30 minutes	613' 2"	**GE** 614'	614' on RVLMS Sensor #8

Refueling Level Gauge Discussion

Although PNP has a refueling level gauge, this instrument cannot be credited. In both cold shutdown and refueling modes, this instrument is not in service except during PCS drain and fill evolutions, and for shiftly log keeping. It is removed from service after each of these evolutions to eliminate a potential PCS inventory loss from accidentally breaking the gauge's glass tube. In either mode, regardless of whether the refueling level gauge

was in service or not at the onset of a loss of PCS inventory, it would become unavailable, as one of the first actions prescribed by procedure during a loss of PCS inventory is to evacuate and close containment.

Alert Deviation Discussion

When reactor vessel water level lowers to the 616' 5.5" elevation, the bottom of the PCS hot leg is uncovered. Reactor vessel level transmitter, LT-0105, and reactor hot leg level transmitter, LT-0106 provide the closest readily available indication of this level. Both transmitters will indicate approximately 0% at a level of 617' 0". This deviation is appropriate for PNP, as it provides equivalent protection of the public health and safety in the absence of an installed reactor vessel level indication that concisely meets the NEI EAL of the "bottom of the RCS loop." Additionally, the level selected is discriminatory and allows escalation to a Site Area Emergency. Therefore, use of this indication does not decrease the overall effectiveness of the EAL scheme.

Site Area Emergency Deviation Discussion

When reactor vessel water level lowers to the 615' 11.5" elevation, level is 6" below the bottom ID of the PCS hot leg. RVLMS is unable to distinguish 6" below the bottom ID of the PCS loop penetration. The closest indication of this level is provided by RVLMS upper guide structure (UGS) region sensor #7 red light on, ~40" above the fuel alignment plate at the 616' 6" elevation. This indication allows clear escalation both from CA1/CA2 to CS1/CS2 and from CS1/CS2 to CG1. It also maintains the 6" difference in PCS elevation between the Alert and Site Area Emergency classification prescribed by NEI 99-01, Revision 4. This deviation is appropriate for PNP, as it provides equivalent protection of the public health and safety in the absence of an installed reactor vessel level indication that concisely meets the NEI EAL of "6" below the bottom of the RCS loop." Additionally, the level selected is discriminatory and allows escalation to a General Emergency. Therefore, use of this indication does not decrease the overall effectiveness of the EAL scheme.

General Emergency Deviation Discussion

"614 ft. 0 in. el." is ~11 in. above the TOAF, and is the lowest level indication that can be used to determine reactor vessel inventory loss at PNP. When reactor vessel water level drops to the 613' 2" elevation (TOAF) [Ref. 2], core uncovery is about to occur. The closest indication of this level is provided by RVLMS UGS region sensor #8 red light on, ~11" above the fuel alignment plate, at the 614' 0" elevation. This deviation is appropriate for PNP, as it provides equivalent protection of the public health and safety in the absence of an installed reactor vessel level indication that concisely meets the NEI EAL of "less than TOAF for > 30 minutes." Additionally, the level selected is the lowest indication available that would allow escalation to a General Emergency. Therefore, use of this indication does not decrease the overall effectiveness of the EAL scheme.

Optional Deviations Discussion

RVLMS sensor #6 indication at the 619' 2" elevation is too high to escalate from the Unusual Event to the Alert classification without the potential of needlessly activating offsite agencies.

RVLMS sensor #7 at the 616' 6" elevation is only 0.5" above the NEI EAL for an Alert at 616' 5.5". If this indication is used for the Alert classification, the only remaining options would be to declare either a Site Area Emergency at 614', RVLMS sensor #8, and deviate from the NEI EAL escalation scheme by having no General Emergency EAL for level, or declare a General Emergency and deviate by having no Site Area Emergency EAL for level. The NEI 99-01 CS1 and CS2 Basis also states "If a PWRs RVLIS is unable to distinguish 6" below the bottom ID of the RCS loop penetration, then the first observable point below the bottom ID of the loop should be chosen as the setpoint." Complying with this would also cause an escalation to a Site Area Emergency at the 614' elevation with no General Emergency EAL for level.

All of these options would decrease the effectiveness of the NEI escalation scheme.

Conclusion

There are only two indications available upon which to base an Alert classification near the NEI EAL, and still have indications left that could be used to escalate the event. Both of those indications (LT-0105 and LT-0106) will trend off-scale low at the 617' elevation. Below the 617' elevation there are only two additional indications (RVLMS sensors #7 and #8) upon which to base the Site Area Emergency and the General Emergency classifications near the NEI EALs. All three of these EAL classifications maintain a level differential that compares favorably to the NEI EALs for escalation of the loss of PCS inventory event. There are 6 inches between the Alert and Site Area Emergency for both the PNP and the NEI EALs. There are 2' 6" between the Site Area Emergency and the General Emergency for the PNP EAL and 2' 9.5" between the Site Area Emergency and the General Emergency for the NEI EALs.

Therefore, PNP's EAL scheme for loss of inventory in cold shutdown and refueling modes will result in the following:

1. Alert Classification: using indications available on LT-0105/0106 at a level of 617' (6.5" above the bottom ID of the RCS loop).
2. SAE Classification:
    a. use the level of 616' 6" for 6" below bottom ID of the RCS loop (0.5" above bottom ID of the RCS loop and therefore 6.5" greater than the NEI setpoint).
    b. use the level of 614' for TOAF (~11" above the TOAF and ~11" greater than the NEI setpoint).
3. GE Classification: use the level of 614' for TOAF (~11" greater than the NEI setpoint).

The PNP site-specific levels used meet the intent of the NEI guidance with regard to loss of inventory with irradiated fuel in the reactor vessel. Therefore, these deviations are appropriate for PNP as they provide equivalent protection of the public health and safety in the absence of installed reactor vessel level indication that concisely meets the NEI EAL. These indications maintain an escalation strategy comparable to the NEI EALs, and therefore do not decrease the effectiveness of the EAL scheme.

Justification Matrix
Addendum #2

Abnormal Radiological Levels / Radiological Effluent

EAL RA1.2 Deviation Justification

Deviation

This deviation relates to NEI 99-01 revision 4, EAL AA1 Example EAL #2, which is PNP EAL RA1.2. The deviation applies specifically to the Radioactive Gaseous Effluent Monitors (RGEM) RIA-2326 and RIA-2327. The deviation is needed to maintain consistency between RU1.2, RA1.2, RS1.1 and RG1.1. The NEI guidance for RA1.2 is a release that "exceeds 200 times the Radiological Technical Specifications". The NEI guidance for RS1.1 is an offsite dose that "exceeds 100 mRem TEDE **OR** 500mRem Thyroid CDE ". When the calculations are done for 200X our Offsite Dose Calculation Manual, the projected offsite dose is 123 mRem using average meteorology and 437 mRem using adverse meteorology. Both of these exceed the Site Area Emergency limit of 100 mRem. Therefore if the 200X ODCM value is used, an Alert would never be declared since the SAE threshold was already exceeded.

The attached comparison of the NEI to PNP Emergency Action Levels for radiological releases provides a comparison of the limits used in EALs RU1.2, RA1.2, RS1.1 and RG1.1.

Monitor Description

RIA-2326 is the normal range noble gas monitor and RIA-2327 is the high range noble gas monitor. RIA-2326 has an "Alert" alarm at 1.6 E+4 CPM and a "High" alarm at 1.3 E+6 CPM. RIA-2326 and RIA-2327 are interlocked such that the High Alarm on RIA-2326 causes the normal sample flow path to be bypassed and sample flow diverted to the high range monitor RIA-2327.

Conclusion

While the limits for RIA-2326 and RIA-2327 deviate from the specific guidance in NEI 99-01, they are consistent with the intent to provide a graduated escalation from the UE to the General Emergency. The proposed scheme will result in an earlier classification at the Alert level but the classification at the Site Area Emergency and General Emergency levels will remain consistent with the NEI 99-01 guidance. Therefore there is no decrease in effectiveness and this scheme continues to protect the health and safety of the public.

NEI EAL#	PNP EAL#	NEI 99-01 Limit	Monitor Values Corresponding to NEI 99-01 Limit	PNP EAL Limit	Basis
AU1 Example 2	RU1.2	2 X ODCM	RIA–2326 3.2 E+5 CPM RIA–2327 N/A	RIA–2326 3.2 E+5 CPM RIA–2327 N/A	No deviation.
AA1 Example 2	RA1.2	200 X ODCM	RIA–2326 3.2 E+7 CPM RIA–2327 28 mRem/hr	RIA–2326 1.3 E+6 CPM RIA–2327 1.5 mRem/hr	The limits for RIA-2326 and RIA-2337 are well below the 200X the ODCM limit. These limits were selected to ensure adequate separation from the limits in RS1.1. The Site Area Emergency limits are based on EPA guidance and use adverse meteorology while the ODCM values used in RA1.2 use average meteorology. The limits correspond to the High alarm on RIA-2326, which is also the point at which the RGEM monitors switch from low range to high range. While the limits for RIA-2326 and RIA-2327 deviate from the specific guidance in NEI 99-01, they are consistent with the intent to provide a graduated escalation from the UE to the General Emergency.
AS1 Example 1	RS1.1	100 mRem TEDE OR 500 mRem Thyroid CDE	RIA–2326 N/A RIA–2327 5.2 mRem/hr	RIA–2326 N/A RIA–2327 5.2 mRem/hr	No deviation.
AG1 Example 1	RG1.1	1000 mRem TEDE OR 5000 mRem Thyroid CDE	RIA–2326 N/A RIA–2327 52 mRem/hr	RIA–2326 N/A RIA–2327 52 mRem/hr	No deviation.

ATTACHMENT 4

EMERGENCY PLAN CHANGES/WALL CHARTS

2 Wall Charts Follow

THIS PAGE IS AN
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FIGURE,
THAT CAN BE VIEWED AT
THE
RECORD TITLED:

"Palisades Nuclear Plant Emergency
Action Level Matrix
El-1, Emergency Classification and
Actions",
(Cold Conditions
PCS≤200 degrees F)

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D-01

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"Palisades Nuclear Plant Emergency
Action Level Matrix
El-1, Emergency Classification and
Actions",
(Hot Conditions -
PCS>200 degrees F)

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D-02

ENCLOSURE 5

CDs OF ENCLOSURES, REFERENCES, AND SUPPORTING DOCUMENTATION

2 Compact Disks Enclosed