Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth. MA 02360

February 18 , 2016

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

SUBJECT: Emergency Plan Implementing Procedure Revisions

Pilgrim Nuclear Power Station
Docket No. 50-293
Renewed License No. DPR-35

LETTER NUMBER: 2.16.013

Dear Sir or Madam:

In accordance with 10 Code of Federal Regulations (CFR) 50.4, Entergy Nuclear Operations, Inc. is providing the latest revision to the Entergy EP-AD-601 and EP-IP-261. The procedure revisions were reviewed in accordance with 10 CFR 50.54(q). The 50.54(q) reviews are also provided in the letter attachment.

The PNPS Emergency Plan continues to meet the planning standards outlined in 10 CFR 50.47, the effectiveness of the emergency plan is not reduced, and the changes did not require prior U.S. Nuclear Regulatory Commission approval.

If you have any questions or require additional information, please contact me at (508) 830-8227.

There are no regulatory commitments contained in this letter.

Sincerely,

Donna Calabrese
Donna Calabrese
Emergency Planning Manager

KLS/jc

Attachment:
1. EP-AD-601, Emergency Action Level Technical Bases Document, Revision 6 and associated 10 CFR 50.54(q) Review
2. EP-IP-261, Technical Support Center (TSC) Operations, Revision 8 and associated 10 CFR 50.54(q) Review.

A X45
NRR

cc:

Director of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

U.S. Nuclear Regulatory Commission
Region 1 – Incident Response Center
2100 Renaissance Blvd., Suite 100
King of Prussia, PA 19406-2713

NRC Resident Inspector
Pilgrim Nuclear Power Station

Attachment 1

Letter Number 2.16.013

EP-AD-601, Emergency Action Level Technical Bases Document, Revision 6 and associated 10 CFR 50.54(q) Review

Emergency Action Level Technical Bases Document

RTYPE H8.26

Change Statement

- Revise Definition of "felt earthquake" to ensure correct interpretation of need for 2 of 3 provided elements to declare the event. Page 70
- Repeat description of "flooding" and reminder not to delay classification while isolating the water source for HU1.4, Page 77.
- Add reminder that fires outside of an H-1 area do not meet the criteria for declaration of HU2.1. Page 99
- Embolden references to Security's role in determining the occurrence of a security event at the site. Pages 115, 116
- Add reinforcement of awareness that temporary lineups to the Torus are not considered a long term heat sink. Page 126
- Add reminder that Judgment classifications should not be used if an applicable EAL classification is applicable. Pages 131, 133, 135, 137, 278, 280, 299, 301, 324 and 326
- Add reinforcement to existing guidance regarding depressurization and restoration using currently available level control measures. Pages 268, 282

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1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Pilgrim Nuclear Power Station (PNPS). It should be used to facilitate review of the PNPS EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EP-IP-100.1, *"Emergency Action Levels (EALs)"*, may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Director in making classifications, particularly those involving judgment or multiple events. The bases information may also be useful in training and for explaining event classifications to offsite officials.

The expectation is that emergency classifications are to be made as soon as conditions are present and recognizable for the classification, but within 15 minutes or less in all cases of conditions present. Use of this document for assistance is not intended to delay the emergency classification.

Attachments 9.9, Abbreviations/Acronyms, and 9.10, PNPS-To-NEI 99-01 EAL Cross-Reference, are to be used to clarify abbreviations and acronyms and as a cross-reference of PNPS EALs to the NEI 99-01 IC/EAL identification scheme.

2.0 REFERENCES

2.1 DEVELOPMENTAL

In addition to the general references listed below, see the EAL basis discussions of Attachments 9.1 through 9.7 for EAL-specific developmental references.

[1] EN-AD-101-01, *"NMM Procedure Writer Manual"*

[2] EP-PP-01, *"PNPS Emergency Plan"*

[3] NEI 99-01 Revision 5 Final, "Methodology for Development of Emergency Action Levels," February 2008

[4] NRC Regulatory Issue Summary (RIS) 2003-18, "Supplement 2, Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels," Revision 4, dated January 2003 (December 12, 2005)

[5] NRC Regulatory Issue Summary (RIS) 2007-01, "Clarification of NRC Guidance for Maintaining a Standard Emergency Action Level Scheme," dated January 10, 2007

[6] PNPS 1.3.4-10, *"Writers' Guide for Emergency Operating Procedures"*

2.2 IMPLEMENTING

[1] EAL wall chart

[2] EP-IP-100, *"Emergency Classification and Notification"*

[3] EP-IP-100.1, *"Emergency Action Levels (EALs)"*

3.0 DEFINITIONS

[1] 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 Specifications LCOs in effect.

(a) 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".

(b) 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".

[2] Bomb - Refers to an explosive device suspected of having sufficient force to damage plant systems or structures.

[3] Civil Disturbance - A group of people violently protesting station operations or activities at the site.

[4] Confinement Boundary - The barrier(s) between areas containing radioactive substances and the environment.

[5] Containment Closure - The action taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment closure is established when Primary or Secondary Containment integrity is established in accordance with Section 3.7 of the Technical Specifications.

[6] Explosion - 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.

[7] Extortion - An attempt to cause an action at the station by threat of force.

[8] Fire - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is not required if large quantities of smoke and heat are observed.

[9] Hostage - Person(s) held as leverage against the station to ensure that demands will be met by the station.

[10] Hostile Action - An act toward PNPS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate plant personnel to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included.

(a) Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on PNPS. Nonterrorism-based EALs should be used to address such activities (e.g., violent acts between individuals in the Owner Controlled Area).

[11] 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.

[12] Imminent - Mitigation actions have been ineffective, additional actions are not expected to be successful, and trended information indicates that the event or condition will occur. Where imminent time frames are specified, they shall apply.

[13] Intrusion - The act of entering without authorization. Discovery of a bomb in a specified area is indication of intrusion into that area by a hostile force.

[14] 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.

[15] Owner Controlled Area - For purpose of EAL classification, the "Owner Controlled Area" is the Entergy owned property on the north side of Rocky Hill Road.

[16] Projectile - An object directed toward an NPP that could cause concern for its continued operability, reliability, or personnel safety.

[17] Protected Area - An area which normally encompasses all controlled areas within the security protected area fence as delineated in PNPS 1.3.131, *"Owner Controlled Area (OCA) Access"*.

[18] Sabotage - Deliberate damage, misalignment, or misoperation 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.

[19] Security Condition - Any security event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A security condition does not involve a hostile action.

[20] Significant Transient - An unplanned event involving any of the following:

(a) Runback > 25% thermal power

(b) Electrical load rejection > 25% full electrical load

(c) Reactor Scram

(d) ECCS injection

(e) Thermal power oscillations > 10%

[21] Strike Action - Work stoppage within the Protected Area by a body of workers to enforce compliance with demands made on PNPS. The strike action must threaten to interrupt Normal Plant Operations.

[22] Unisolable - A breach or leak that cannot be promptly isolated.

[23] Unplanned - A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

[24] Valid - An indication, report, or condition is considered to be valid when it is verified by:

(a) an instrument channel check; or

(b) indications on related or redundant indicators; or

(c) by direct observation by plant personnel such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

[25] Visible Damage - 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.

[26] Vital Areas - 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.

4.0 RESPONSIBILITIES

The Emergency Planning Manager shall be responsible for:

[1] Ensuring that each EAL listed in EP-IP-100.1, *"Emergency Action Levels (EALs)"* has a technically sound basis described within Attachments 9.1 through 9.7 of this Procedure.

[2] Ensuring that, for any proposed revision to an EAL listed in EP-IP-100.1, *"Emergency Action Levels (EALs)"*:

(a) A technical basis has been developed to support the proposed revision.

(b) The technical basis is included as part of the revision review and approval processes.

(c) Changes are performed and documented in accordance with the plant administrative procedures which include evaluating changes in accordance with 10CFR50.54(q) requirements

[3] Ensuring that any approved changes to an EAL listed in EP-IP-100.1, *"Emergency Action Levels (EALs)"* are reflected on the controlled copies of the EAL wall charts and EAL Technical Bases Document.

5.0 DETAILS

5.1 PRECAUTIONS AND LIMITATIONS

None

5.2 PROCEDURE

[1] The Emergency Planning Manager shall, upon review of a proposed revision to one or more EALs listed in EP-IP-100.1, *"Emergency Action Levels (EALs)"*, verify that:

(a) A technical basis has been developed to support the proposed revision.

(b) The technical basis is included as part of the revision review and approval processes.

(c) Changes are performed and documented in accordance with the plant administrative procedures which include evaluating changes in accordance with 10CFR50.54(q) requirements.

(d) If the proposed change affects the wording of the EAL wall charts, a revision to the wall charts is made prior to implementation of the revision.

[2] The Emergency Planning Manager shall, upon review of any proposed revision to the EALs listed in EP-IP-100.1, *"Emergency Action Levels (EALs)"*, ensure that changes are performed and documented in accordance with plant administrative procedures. This includes evaluating changes in accordance with 10CFR50.54(q) requirements for the applicable source documents referenced and listed in this procedure and for any potential impact based on those EAL technical bases which reference them.

6.0 INTERFACES

Bechtel Drawing Electrical single line diagram S-E-155

BWROG EPG/SAG Revision 2, Sections PC/G

EOP-01, *"RPV Control"*

EOP-01, *"RPV Control"*, Entry Condition

EOP-02, *"RPV Control, Failure-To-Scram"*

EOP-03, *"Primary Containment Control"*

EOP-03, *"Primary Containment Control"*, Entry Condition

EOP-04, *"Secondary Containment Control"*

EOP-05, *"Radioactive Release Control"*

EOP-18, *"Steam Cooling"*

EOP-16, *"RPV Flooding"*

EOP-17, *"Emergency RPV Depressurization"*

EOP-26, *"RPV Flooding, Failure-to-Scram"*

EOP-27, *"Emergency RPV Depressurization, Failure-to-Scram"*

EP-AD-413, *"Emergency Communications Test"*

EP-IP-300, *"Offsite Radiological Dose Assessment"*

EP-IP-310, *"Offsite Monitoring Team Activation And Response"*

FSAR Figures 5.2-1 through 5.2-6

FSAR Figure 8.6-1

FSAR Sections 4.4, 4.5, 4.6, 4.11 (MSL)

FSAR Section 4.7 (RCIC)

FSAR Section 4.9 (RWCU)

FSAR Section 4.10 - Nuclear System Leakage Rate Limits

FSAR Section 5.2 - Primary Containment System

FSAR Section 5.3 - Secondary Containment System

FSAR Section 6.3 (HPCI)

FSAR Section 8.1

FSAR Section 8.3

FSAR Section 8.5

FSAR Section 9.2 - Liquid Radwaste System

FSAR Section 10.15

FSAR Section 11.9 (FW)

FSAR Table 5.2-1

Modification ER03114340

NEI 99-01 Definitions and Appendix C

NEI 99-01 Revision 5

NEI/NRC EAL FAQ #2006-014

NEI White Paper, "Enhancements to Emergency Preparedness Programs for Hostile Action", November 18, 2005

NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security Based Events", July 18, 2005

PNPS 1.3.40, *"Specifications For Vital Area Barrier Openings, Degradation, And Repair"*

PNPS 1.4.12, *"Primary Containment Entry"*

PNPS 2.1.6, *"Reactor Scram"*

PNPS 2.1.7, *"Vessel Heatup and Cooldown"*

PNPS 2.2.120, *"Postaccident Monitoring Panel"*

PNPS 2.2.126, *"Anticipated Transient Without Scram (ATWS)"*

PNPS 2.2.133, *"H2/O2 Analyzer and C19 Systems"*

PNPS 2.2.14, *"125V DC Battery Systems"*

PNPS 2.2.146, *"Station Blackout Diesel Generator"*

PNPS 2.2.17, *"Communications Systems"*

PNPS 2.2.21, *"High Pressure Coolant Injection System (HPCI)"*

PNPS 2.2.22, *"Reactor Core Isolation Cooling System (RCIC)"*

PNPS 2.2.32, *"Salt Service Water System (SSW)"*

PNPS 2.2.62, *"Area Radiation Monitoring System"*

PNPS 2.2.64, *"Source Range Monitoring System"*

PNPS 2.2.77, *"Drywell Leak Detection Systems"*

PNPS 2.2.79, *"Reactor Protection System"*

PNPS 2.2.8, *"Standby AC Power System (Diesel Generators)"*

PNPS 2.2.80, *"Reactor Vessel Level, Temperature, And Internal Pressure Instrumentation"*

PNPS 2.2.83, *"Reactor Cleanup System"*

PNPS 2.2.85, *"Fuel Pool Cooling and Filtering System"*

PNPS 2.2.92, *"Main Steam Line Isolation and Turbine Bypass Valves"*

PNPS 2.2.94, *"Seawater System"*

PNPS 2.3.1, *"General Action for Alarm Response and Annunciator Control"*

PNPS 2.4.143, *"Shutdown from Outside Control Room"*

PNPS 2.4.144, *"Degraded Voltage"*

PNPS 2.4.154, *"Intake Structure Fouling"*

PNPS 2.4.155, *"Loss of Annunciator System"*

PNPS 2.4.25, *"Loss of Shutdown Cooling"*

PNPS 2.4.31, *"Reactor Basin and/or Spent Fuel Pool Drain-Down"*

PNPS 2.4.44, *"Loss of Drywell Area Coolers"*

PNPS 2.4.57, *"Loss of Public-Address System"*

PNPS 2.4.A.23, *"Loss/Degradation Of 23kV Line"*

PNPS 2.4.A.5, *"Loss of Electrical Bus A5"*

PNPS 2.4.A.6, *"Loss of Electrical Bus A6"*

PNPS 2.5.2.71, *"Radwaste Collection System"*

PNPS 3.M.2-5.1, *"Source Range Monitor Calibration Instruction"*

PNPS 3.M.2-40, *"Refuel Outage Temporary Modification Reactor Shutdown/Flood-Up Level Indication"*

PNPS 5.2.1, *"Earthquake"*

PNPS 5.3.11, *"Loss of Essential DC Bus D16 or D4 and D36"*

PNPS 5.3.12, *"Loss of Essential DC Bus D17 or D5 and D37"*

PNPS 5.3.31, *"Station Blackout"*

PNPS 5.4.3, *"Refueling Floor High Radiation"*

PNPS 5.5.1, *"General Fire Procedure"*

PNPS 5.5.2, *"Special Fire Procedure"*

PNPS 5.5.4, *"Response to Hazardous Material Incidents"*

PNPS 6.3-064, *"Routine Radiological Surveillance Program"*

PNPS 6.5-160, *"Calibration of the Area Radiation Monitoring System"*

PNPS 8.A.13, *"Plant Emergency Alarms and Radio Test"*

PNPS 8.E.29, *"Salt Service Water System Instrumentation Calibration"*

PNPS 8.M.2-6.1, *"Reactor Pressure Readout"*

PNPS ARP-C3RC-A7

PNPS ARP-C3RC-B7

PNPS ARP-C904LC-A4

PNPS ARP-C904LC-B4

PNPS ARP-C904LC-B7

PNPS ARP-C904LC-C7

PNPS Emergency Plan Section B-3

PNPS NE-07-00006 Rev. 0

PNPS ODCM Section 7.1.1, Liquid Radioactive Waste Effluent Release

PNPS ODCM Section 7.2.1, Liquid Radioactive Waste Effluent Monitoring System

PNPS ODCM Section 7.2.2, Main Stack Gas Monitoring System

PNPS ODCM Section 7.2.3, Reactor Building Exhaust Vent Monitor System

PNPS ODCM Section 8.1, Liquid Effluent Monitor

PNPS ODCM Section 8.3, Steam Jet Air Ejector Monitor

PNPS ODCM Table 4.2-1, Radioactive Liquid Waste Sampling and Analysis Program

PNPS ODCM Table 4.3-1, Radioactive Gaseous Waste Sampling and Analysis Program

PNPS Plant-Specific and Severe Accident Management Guidelines

PNPS Technical Specifications

PNPS Technical Specifications, 1.0 Definitions

PNPS Technical Specifications Section 3.1

PNPS Technical Specifications Section 3.14

PNPS Technical Specifications Section 3.6.B

PNPS Technical Specifications Section 3.6.C

PNPS Technical Specifications Section 3.7

PNPS Technical Specifications Section 3.7.C

Radiological Gaseous Effluent EAL Values (EALs AG1.1, AS1.1, AA1.1 and AU1.1) Revision 0, Prepared By: Scott McCain (05/5/09), Reviewed By: Ed Salomon (05/15/09) and Approved By: Kevin Wolf (05/15/09)

SAG-02, *"Containment and Radioactivity Release Control"*

SUDDSRF96-35, Wind and Tornado Evaluation for PNPS

7.0 RECORDS

This document does not generate any records.

8.0 REQUIREMENTS AND COMMITMENTS

This section lists those external commitments (NRC commitments, QA audit findings, and INPO inspection items) implemented in this Procedure.

Reference Document	Commitment	Affected Section(s)/Step(s)
NRC Inspection Finding 81-15-34	Develop and implement a system for use by the Control Room staff to aid in promptly classifying events.	Attachments 9.1 through 9.7. (See also EAL wall chart)
NRC Inspection Finding 81-15-35	Provide EALs which include specific and observable Control Room instrument readings for each EAL corresponding to the respective initiating condition	Attachments 9.1 through 9.7. (See also EAL wall chart)
NRC Inspection Finding 84-05-04	Provide EALs based on field monitoring results and on the methods used if the effluent and containment monitors are inoperable or off-scale.	Attachment 9.1 (EALS AG1.3 and AS1.3, AG1.2 and AG1.3)

Reference Document	Commitment	Affected Section(s)/Step(s)
NRC Inspection Report 50-293/88-28 Item 2.4	Provide further clarification and quantification for earthquake EALs.	Attachment 9.2 (EAL HA1.1, HU1.1)
NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security Based Events"	Revise ECL criteria definitions and subcategory security threat EALs to include expanded security based events.	Attachment 9.2 (EALs HG4.1, HS4.1, HA4.1, HU4.1)

9.0 ATTACHMENTS

9.1 CATEGORY A, ABNORMAL RAD RELEASE/RAD EFFLUENT

9.2 CATEGORY H, HAZARDS

9.3 CATEGORY S, SYSTEM MALFUNCTION

9.4 CATEGORY F, FISSION PRODUCT BARRIER DEGRADATION

9.5 CATEGORY C, COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

9.6 FISSION PRODUCT BARRIER LOSS/POTENTIAL LOSS MATRIX AND BASES

9.7 CATEGORY E, ISFSI

9.8 BACKGROUND AND DISCUSSION

- 9.8.1 Background
- 9.8.2 Fission Product Barriers
- 9.8.3 Emergency Classification Based on Fission Product Barrier Degradation
- 9.8.4 EAL Relationship to EOPs
- 9.8.5 Symptom-Based vs. Event-Based Approach
- 9.8.6 EAL Organization
- 9.8.7 Technical Bases Information

Category A - Abnormal Rad Release/Rad Effluent

EAL Group: ANY (EALs in this category are applicable to any plant condition, hot or cold.)

Many EALs are based on actual or potential degradation of fission product barriers because of the elevated potential for offsite radioactivity release. Degradation of fission product barriers though is not always apparent via nonradiological symptoms. Therefore, direct indication of elevated radiological effluents or area radiation levels are appropriate symptoms for emergency classification.

At lower levels, abnormal radioactivity releases may be indicative of a failure of containment systems or precursors to more significant releases. At higher release rates, offsite radiological conditions may result which require offsite protective actions. Elevated area radiation levels in plant may also be indicative of the failure of containment systems or preclude access to plant vital equipment necessary to ensure plant safety.

Events of this category pertain to the following subcategories:

1. Offsite Rad Conditions

Direct indication of effluent radiation monitoring systems provides a rapid assessment mechanism to determine releases in excess of classifiable limits. Projected offsite doses, actual offsite field measurements, or measured release rates via sampling indicate doses or dose rates above classifiable limits.

2. Onsite Rad Conditions & Spent Fuel Pool Events

Sustained general area radiation levels in excess of those indicating loss of control of radioactive materials also warrant emergency classification.

3. MCR/CAS Radiation

Sustained general area radiation levels in excess of those levels which may preclude access to vital plant areas also warrant emergency classification

Unusual Event - AU1.1

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Offsite Rad Conditions

Initiating Condition: **Any** release of gaseous or liquid radioactivity to the environment greater than 2 times the Radiological Effluent Technical Specifications/ODCM for 60 minutes or longer

EAL:

AU1.1 Unusual Event

Any valid gaseous monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)

Note 2: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Unusual Event - AU1.1

Table A-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	Stack Gas	RM-1705-18A/B (Panel C910 - units of cps) Low Range	-----	-----	4E+5 cps	2E+4 cps
GASEOUS	Stack Gas	RI-1001-608 (Panel C170 - units of R/hr) High Range	20 R/hr	2 R/hr	-----	-----
GASEOUS	Rx Bldg Vent Exhaust	RM-1705-32A/B (Panel C910 - units of cps)	-----	-----	1E+5 cps	1E+3 cps
LIQUID	Radwaste Discharge Effluent	RM-1705-30 (Panel C910 - units of cps)	-----	-----	200 x hi-hi alarm* not to exceed 8E+5 cps	2 x hi-hi alarm*

* with Radwaste discharge **not** isolated

Mode Applicability:

All

NEI 99-01 Basis:

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses a potential 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.

Unusual Event - AU1.1

Nuclear power plants incorporate 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. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

PNPS has found it advantageous to address gaseous and liquid releases with separate EALs.

The ODCM multiples are specified in EALs AU1.1/AU1.2 and AA1.1/AA1.2 only to distinguish between nonemergency 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.

This EAL 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.

This EAL addresses radioactivity releases that, for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the IC.

This EAL is intended for sites that have established effluent monitoring on nonroutine release pathways for which a discharge permit would not normally be prepared.

EALs AU1.1 and AU1.2 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 setpoints.

The underlying basis of this EAL involves the degradation in the level of safety of the plant implied by the uncontrolled release.

Unusual Event - AU1.1

PNPS Basis:

Gaseous releases in excess of two times the PNPS Offsite Dose Calculation Manual (ODCM) instantaneous limits that continue for greater than 60 minutes represent an uncontrolled situation and, hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

The radiation monitors that detect gaseous radioactivity effluent release to the environment are the Main Stack process radiation monitors RM-1705-18A/B and the Reactor Building Ventilation Exhaust (RBVE) monitors RM-1705-32A/B (ref. 1 and 2).

The Main Stack process radiation monitor system initiates alarms whenever stack radioactivity release levels approach unacceptable limits. The Main Stack directs the effluents from one or more of the following systems to an elevated filtered release point:

- Standby Gas Treatment System
- Augmented Offgas System
- Condenser Air Removal System Mechanical Vacuum Pump
- Gland Seal Exhauster

Indication of Main Stack effluent is provided by process radiation monitors RM-1705-18A/B on Panel C910 and recorder 40-RR-1705-19 on Panel C902 (ref. 1). The RBVE monitor alarms whenever effluent radioactivity release levels approach unacceptable limits. Air from areas containing potential sources of radioactive contamination such as the Reactor Building, Radwaste Building basement, and Turbine Building basement are discharged through the Reactor Building exhaust vent.

Indication of RBV effluent is provided by process radiation monitors RM-1705-32A/B on Panel C910 and recorder 40-RR-1705-19 on Panel C902 (ref. 2).

Unusual Event - AU1.1

Complete assumptions and inputs for these EAL threshold values are documented from the calculation in Radiological Gaseous Effluent EAL Values (ref. 3). The calculated Main Stack and RBV values were rounded down to the most readable division marking on the limiting monitor for the EAL thresholds.

PNPS Basis Reference(s):

1. PNPS ODCM Section 7.2.2, Main Stack Gas Monitoring System
2. PNPS ODCM Section 7.2.3, Reactor Building Exhaust Vent Monitor System
3. Radiological Gaseous Effluent EAL Values (EALs AG1.1, AS1.1, AA1.1, and AU1.1) Revision 0, Prepared By: Scott McCain (05/5/09), Reviewed By: Ed Salomon (05/15/09) and Approved By: Kevin Wolf (05/15/09)

Unusual Event - AU1.2

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Offsite Rad Conditions

Initiating Condition: **Any** release of gaseous or liquid radioactivity to the environment greater than 2 times the Radiological Effluent Technical Specifications/ODCM for 60 minutes or longer

EAL:

AU1.2 Unusual Event

Valid radwaste effluent radiation monitor RM-1705-30 (Panel C910) reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)

AND

Radwaste discharge is **not** isolated

Note 2: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Unusual Event - AU1.2

Table A-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	Stack Gas	RM-1705-18A/B (Panel C910 - units of cps) Low Range	-----	-----	4E+5 cps	2E+4 cps
GASEOUS	Stack Gas	RI-1001-608 (Panel C170 - units of R/hr) High Range	20 R/hr	2 R/hr	-----	-----
GASEOUS	Rx Bldg Vent Exhaust	RM-1705-32A/B (Panel C910 - units of cps)	-----	-----	1E+5 cps	1E+3 cps
LIQUID	Radwaste Discharge Effluent	RM-1705-30 (Panel C910 - units of cps)	-----	-----	200 x hi-hi alarm* not to exceed 8E+5 cps	2 x hi-hi alarm*

* with Radwaste discharge **not** isolated

Mode Applicability:

All

NEI 99-01 Basis:

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses a potential 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.

Unusual Event - AU1.2

Nuclear power plants incorporate 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. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

PNPS has found it advantageous to address gaseous and liquid releases with separate EALs.

The ODCM multiples are specified in EALs AU1.1/AU1.2 and AA1.1/AA1.2 only to distinguish between nonemergency 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.

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the IC established by the radioactivity discharge permit. This value may be associated with a planned batch release, or a continuous release path.

EAL AU1.1 and EAL AU1.2 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 setpoints.

The underlying basis of this EAL involves the degradation in the level of safety of the plant implied by the uncontrolled release.

PNPS Basis:

Liquid releases in excess of two times the Hi-Hi alarm that continue for greater than 60 minutes represent an uncontrolled situation and, hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

Unusual Event - AU1.2

Indication of radwaste effluent is provided by process radiation monitor RM-1705-30 on Panel C910 and recorder RR-1792 (red pen) on Panel C20 (Radwaste Control Room). The Radwaste Effluent Radiation Monitoring System monitors radwaste discharges to the discharge canal and provides alarm and automatic isolation functions if radioactivity levels exceed predetermined setpoints. The alarm and isolation setpoints are calculated for each discharge to ensure ODCM liquid effluent limits are not exceeded (ref. 1, 2).

At low classification levels, the concern for classification is the continuing, uncontrolled release of radioactivity and not the magnitude of the release. When the liquid release is isolated, the release is no longer continuing nor is it uncontrolled. Therefore, the classification is not appropriate when the liquid release is isolated. Radwaste effluent discharge isolation valves FV-7214A and B close if radwaste effluent radiation levels exceed the Hi-Hi alarm setpoint (ref. 3).

PNPS Basis Reference(s):

1. PNPS ODCM Section 7.1.1, Liquid Radioactive Waste Effluent Release
2. PNPS ODCM Section 8.1, Liquid Effluent Monitor
3. PNPS ODCM Section 7.2.1, Liquid Radioactive Waste Effluent Monitoring System

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Offsite Rad Conditions

Initiating Condition: **Any** release of gaseous or liquid radioactivity to the environment greater than 2 times the Radiological Effluent Technical Specifications/ODCM for 60 minutes or longer

EAL:

AU1.3 Unusual Event

Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x ODCM limits for ≥ 60 min. (Note 2)

Note 2: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Mode Applicability:

All

NEI 99-01 Basis:

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses a potential 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.

Unusual Event - AU1.3

Nuclear power plants incorporate 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. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

PNPS has found it advantageous to address gaseous and liquid releases with separate EALs.

The ODCM multiples are specified in EALs AU1.3 and AA1.3 only to distinguish between nonemergency 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.

This EAL 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.

This EAL 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 water systems, etc.

The underlying basis of this EAL involves the degradation in the level of safety of the plant implied by the uncontrolled release.

PNPS Basis

Confirmed sample analyses in excess of two times the PNPS Offsite Dose Calculation Manual (ODCM) instantaneous limits that continue for greater than 60 minutes represent an uncontrolled situation and, hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. This EAL addresses collecting and analyzing both gaseous and liquid effluent samples to ensure that release conditions above nominal steady state conditions are detected and reported (ref. 1, 2).

Unusual Event - AU1.3

PNPS Basis Reference(s):

1. PNPS ODCM Table 4.2-1, Radioactive Liquid Waste Sampling and Analysis Program
2. PNPS ODCM Table 4.3-1, Radioactive Gaseous Waste Sampling and Analysis Program

Alert - AA1.1

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Offsite Rad Conditions

Initiating Condition: **Any** release of gaseous or liquid radioactivity to the environment greater than 200 times the Radiological Effluent Technical Specifications/ODCM for 15 minutes or longer

EAL:

AA1.1 Alert

Any valid gaseous monitor reading > Table A-1 column "Alert for ≥ 15 min." (Note 2)

Note 2: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Alert - AA1.1

Table A-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	Stack Gas	RM-1705-18A/B (Panel C910 - units of cps) Low Range	----	----	4E+5 cps	2E+4 cps
GASEOUS	Stack Gas	RI-1001-608 (Panel C170 - units of R/hr) High Range	20 R/hr	2 R/hr	----	----
GASEOUS	Rx Bldg Vent Exhaust	RM-1705-32A/B (Panel C910 - units of cps)	----	----	1E+5 cps	1E+3 cps
LIQUID	Radwaste Discharge Effluent	RM-1705-30 (Panel C910 - units of cps)	----	----	200 x hi-hi alarm* not to exceed 8E+5 cps	2 x hi-hi alarm*

* with Radwaste discharge **not** isolated

Mode Applicability:

All

NEI 99-01 Basis:

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential 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 incorporate 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. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

PNPS has found it advantageous to address gaseous and liquid releases with separate EALs.

The ODCM multiples are specified in EALs AU1.1/AU1.2 and AA1.1/AA1.2 only to distinguish between nonemergency 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.

This EAL 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.

This EAL is intended for sites that have established effluent monitoring on nonroutine release pathways for which a discharge permit would not normally be prepared.

The underlying basis of this EAL involves the degradation in the level of safety of the plant implied by the uncontrolled release.

PNPS Basis:

This event escalates from the Unusual Event by escalating the magnitude of the release by a factor of 100 for the RBV. A multiple of 200 times the ODCM limit is not used for the Main Stack EAL as it would result in the Alert value being higher than the Site Area Emergency value. Therefore, to provide a numeric progression for the EAL, the AA1.1 value is set logarithmically half way between the AS1.1 value and the AU1.1 value. The radiation monitors that detect gaseous radioactivity effluent release to the environment are the Main Stack process radiation monitors RM-1705-18A/B and the Reactor Building Ventilation Exhaust (RBVE) monitors RM-1705-32A/B on Panel C910 (ref .1, 2).

Complete assumptions and inputs for these EAL threshold values are documented from the calculation in Radiological Gaseous Effluent EAL Values (ref. 3). The calculated Main Stack and RBV values were rounded down to the most readable division marking on the limiting monitor for the EAL thresholds.

Alert - AA1.1

PNPS Basis Reference(s):

1. PNPS ODCM Section 7.2.2, Main Stack Gas Monitoring System
2. PNPS ODCM Section 7.2.3, Reactor Building Exhaust Vent Monitor System
3. Radiological Gaseous Effluent EAL Values (EALs AG1.1, AS1.1, AA1.1 and AU1.1) Revision 0, Prepared By: Scott McCain (05/5/09), Reviewed By: Ed Salomon (05/15/09) and Approved By: Kevin Wolf (05/15/09)

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Offsite Rad Conditions

Initiating Condition: **Any** release of gaseous or liquid radioactivity to the environment greater than 200 times the Radiological Effluent Technical Specifications/ODCM for 15 minutes or longer

EAL:

AA1.2 Alert

Valid radwaste effluent radiation monitor RM-1705-30 (Panel C910) reading > Table A-1 column "Alert for ≥ 15 min." (Note 2)

AND

Radwaste discharge is **not** isolated

Note 2: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Alert - AA1.2

Table A-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	Stack Gas	RM-1705-18A/B (Panel C910 - units of cps) Low Range	-----	-----	4E+5 cps	2E+4 cps
GASEOUS	Stack Gas	RI-1001-608 (Panel C170 - units of R/hr) High Range	20 R/hr	2 R/hr	-----	-----
GASEOUS	Rx Bldg Vent Exhaust	RM-1705-32A/B (Panel C910 - units of cps)	-----	-----	1E+5 cps	1E+3 cps
LIQUID	Radwaste Discharge Effluent	RM-1705-30 (Panel C910 - units of cps)	-----	-----	200 x hi-hi alarm* not to exceed 8E+5 cps	2 x hi-hi alarm*

* with Radwaste discharge **not** isolated

Mode Applicability:

All

NEI 99-01 Basis:

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential 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.

Alert - AA1.2

Nuclear power plants incorporate 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. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

PNPS has found it advantageous to address gaseous and liquid releases with separate EALs.

The ODCM multiples are specified in EALs AU1.1/AU1.2 and AA1.1/AA1.2 only to distinguish between nonemergency 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.

This EAL 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.

This EAL addresses radioactivity releases that, for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the IC established by the radioactivity discharge permit. This value may be associated with a planned batch release or a continuous release path.

The underlying basis of this EAL involves the degradation in the level of safety of the plant implied by the uncontrolled release.

PNPS Basis:

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

The threshold of > 200 times the Hi-Hi alarm setpoint is limited to a maximum value of 8E+5 cps to assure an on-scale readable value.

PNPS Basis Reference(s):

1. PNPS ODCM Section 7.1.1, Liquid Radioactive Waste Effluent Release
2. PNPS ODCM Section 8.1, Liquid Effluent Monitor
3. PNPS ODCM Section 7.2.1, Liquid Radioactive Waste Effluent Monitoring System

Alert - AA1.3

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Offsite Rad Conditions

Initiating Condition: **Any** release of gaseous or liquid radioactivity to the environment greater than 200 times the Radiological Effluent Technical Specifications/ODCM for 15 minutes or longer

EAL:

AA1.3 Alert

Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODCM limits for ≥ 15 min. (Note 2)

Note 2: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Mode Applicability:

All

NEI 99-01 Basis:

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential 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.

Alert - AA1.3

Nuclear power plants incorporate 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. 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 EALs AU1.3 and AA1.3 only to distinguish between nonemergency 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.

This EAL 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.

This EAL 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 water systems, etc.

The underlying basis of this EAL involves the degradation in the level of safety of the plant implied by the uncontrolled release.

PNPS Basis:

Confirmed sample analyses in excess of 200 times the PNPS Offsite Dose Calculation Manual (ODCM) limits that continue for greater than 15 minutes represent an uncontrolled situation and, hence, a potential degradation in the level of safety. This event escalates from the Unusual Event by raising the magnitude of the release by a factor of 100 over the Unusual Event level (i.e., 200 times ODCM).

The required release duration was reduced to 15 minutes in recognition of the raised severity.

PNPS Basis Reference(s):

1. PNPS ODCM Table 4.2-1 Radioactive Liquid Waste Sampling and Analysis Program
2. PNPS ODCM Table 4.3-1 Radioactive Gaseous Waste Sampling and Analysis Program

Site Area Emergency - AS1.1

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mrem TEDE or 500 mrem thyroid CDE for the actual or projected duration of the release

EAL:

AS1.1	**Site Area Emergency**
Valid Main Stack High Range Effluent Monitor (RI-1001-608) reading > Table A-1 column "SAE for ≥ 15 min." (Note 1)	

Note 1: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. (See EAL AS1.2.) Do not delay declaration awaiting dose assessment results.

Site Area Emergency - AS1.1

Table A-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	Stack Gas	RM-1705-18A/B (Panel C910 - units of cps) Low Range	----	----	4E+5 cps	2E+4 cps
GASEOUS	Stack Gas	RI-1001-608 (Panel C170 - units of R/hr) High Range	20 R/hr	2 R/hr	----	----
GASEOUS	Rx Bldg Vent Exhaust	RM-1705-32A/B (Panel C910 - units of cps)	----	----	1E+5 cps	1E+3 cps
LIQUID	Radwaste Discharge Effluent	RM-1705-30 (Panel C910 - units of cps)	----	----	200 x hi-hi alarm* not to exceed 8E+5 cps	2 x hi-hi alarm*

* with Radwaste discharge **not** isolated

Mode Applicability:

All

NEI 99-01 Basis:

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

The site specific monitor list in Table A-1 includes effluent monitors on all potential release pathways.

PNPS Basis:

The radiation monitors that detect gaseous radioactivity effluent release to the environment include both the Main Stack process radiation monitors and the Reactor Building Ventilation Exhaust (RBVE) process radiation monitors. In an accident condition or when high radiation condition exists in the Reactor Building, the Reactor Building Isolation System (RBIS) automatically trips and isolates the normal Reactor Building ventilation exhaust system and also starts the Standby Gas Treatment System which then exhausts air from the Reactor Building directly to the Main Stack. The Main Stack release point is therefore the only gaseous release pathway suitable for assessing this EAL (ref. 1).

The Main Stack process radiation monitor system initiates alarms whenever stack radioactivity release levels approach unacceptable limits. The Main Stack directs the effluents from one or more of the following systems to an elevated filtered release point:

- Standby Gas Treatment System
- Augmented Offgas System
- Condenser Air Removal System Mechanical Vacuum Pump
- Gland Seal Exhauster

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is 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 dose assessment results are available, declaration should be based on dose assessment instead of the radiation monitor EAL threshold values. (See EAL AS1.2). However, the emergency declaration should not be delayed awaiting dose assessment results.

Complete assumptions and inputs for these EAL threshold values are documented from the calculation in Radiological Gaseous Effluent EAL Values (ref. 2). The calculated Main Stack value was rounded down to the most readable division marking on the limiting monitor for the EAL threshold.

Site Area Emergency - AS1.1

PNPS Basis Reference(s):

1. PNPS FSAR Section 5.3, Secondary Containment System
2. Radiological Gaseous Effluent EAL Values (EALs AG1.1, AS1.1, AA1.1 and AU1.1) Revision 0, Prepared By: Scott McCain (05/5/09), Reviewed By: Ed Salomon (05/15/09) and Approved By: Kevin Wolf (05/15/09)

Site Area Emergency - AS1.2

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mrem TEDE or 500 mrem thyroid CDE for the actual or projected duration of the release

EAL:

AS1.2 Site Area Emergency

Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or > 500 mrem thyroid CDE at or beyond the site boundary

Mode Applicability:

All

NEI 99-01 Basis:

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is 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 EAL.

Site Area Emergency - AS1.2

PNPS Basis:

The Site Area values are based on doses at or beyond the site boundary that result from an actual or imminent release of gaseous radioactivity that exceeds 100 mrem TEDE or 500 mrem CDE thyroid for the actual or projected duration of the release.

When dose assessment results (using actual meteorology and release information) become available, they will be used to determine whether the integrated dose exceeds the threshold values of > 100 mrem TEDE or > 500 mrem thyroid CDE at or beyond the site boundary. Actual meteorology is specifically identified since it gives the most accurate dose assessment. If the dose assessment results are available at the time that the classification is made, the results will be used in conjunction with this EAL for classifying the event rather than effluent radiation monitor EAL threshold values.

PNPS Basis Reference(s):

1. EP-IP-300, *"Offsite Radiological Dose Assessment"*

Site Area Emergency - AS1.3

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mrem TEDE or 500 mrem thyroid CDE for the actual or projected duration of the release

EAL:

AS1.3 Site Area Emergency

Field survey indicates closed window dose rate > 100 mR/hr expected to continue for ≥ 1 hr at or beyond the site boundary

OR

Field survey sample analysis indicates thyroid CDE > 500 mrem for 1 hr of inhalation at or beyond the site boundary

Mode Applicability:

All

NEI 99-01 Basis:

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Site Area Emergency - AS1.3

PNPS Basis:

Although this EAL initiating condition references the TEDE or thyroid CDE as an integrated dose, field survey results are not generally reported in these dose quantities, but rather in terms of a dose rate or an air sample concentration, respectively. For this reason, the field survey EALs are based on a closed window dose rate greater than 100 mR/hr expected to continue for 1 hour or longer; or analyses of field survey samples indicate thyroid CDE greater than 500 mrem for 1 hour of inhalation at or beyond the site boundary and whichever is more limiting.

PNPS Basis Reference(s):

1. EP-IP-310, *"Radiation Monitoring Team Activation And Response"*

General Emergency - AG1.1

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

AG1.1 General Emergency

Valid Main Stack High Range Effluent Monitor (RI-1001-608) reading > Table A-1 column "GE for ≥ 15 min." (Note 1)

Note 1: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. (See EAL AG1.2.) Do not delay declaration awaiting dose assessment results.

General Emergency - AG1.1

Table A-1 Effluent Monitor Classification Thresholds						
Release Point		**Monitor**	**GE** for ≥ 15 min.	**SAE** for ≥ 15 min.	**ALERT** for ≥ 15 min.	**UE** for ≥ 60 min.
GASEOUS	Stack Gas	RM-1705-18A/B (Panel C910 - units of cps) Low Range	-----	-----	4E+5 cps	2E+4 cps
GASEOUS	Stack Gas	RI-1001-608 (Panel C170 - units of R/hr) High Range	20 R/hr	2 R/hr	-----	-----
GASEOUS	Rx Bldg Vent Exhaust	RM-1705-32A/B (Panel C910 - units of cps)	-----	-----	1E+5 cps	1E+3 cps
LIQUID	Radwaste Discharge Effluent	RM-1705-30 (Panel C910 - units of cps)	-----	-----	200 x hi-hi alarm* not to exceed 8E+5 cps	2 x hi-hi alarm*

* with Radwaste discharge **not** isolated

Mode Applicability:

All

NEI 99-01 Basis:

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

The site specific monitor list in Table A-1 includes effluent monitors on all potential release pathways.

General Emergency - AG1.1

PNPS Basis:

The General Emergency effluent monitor readings are one decade greater than the Site Area Emergency value (refer to EAL AS1.1).

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is 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 dose assessment results are available, declaration should be based on dose assessment instead of the radiation monitor EAL threshold values. (See EAL AG1.2.) However, the emergency declaration should not be delayed awaiting dose assessment results.

Complete assumptions and inputs for these EAL threshold values are documented from the calculation in Radiological Gaseous Effluent EAL Values (ref. 2). The calculated Main Stack value was rounded down to the most readable division marking on the limiting monitor for the EAL threshold.

PNPS Basis Reference(s):

1. PNPS FSAR Section 5.3, Secondary Containment System
2. Radiological Gaseous Effluent EAL Values (EALs AG1.1, AS1.1, AA1.1 and AU1.1) Revision 0, Prepared By: Scott McCain (05/5/09), Reviewed By: Ed Salomon (05/15/09) and Approved By: Kevin Wolf (05/15/09)

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

AG1.2 General Emergency

Dose assessment using actual meteorology indicates doses > 1,000 mrem TEDE or > 5,000 mrem thyroid CDE at or beyond the site boundary

Mode Applicability:

All

NEI 99-01 Basis:

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is 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 EAL.

General Emergency - AG1.2

PNPS Basis:

The General Emergency values are based on doses at or beyond the site boundary that result from an actual or imminent release of gaseous radioactivity that exceeds 1000 mrem TEDE or 5000 mrem CDE thyroid for the actual or projected duration of the release.

When dose assessment results (using actual meteorology and release information) become available, they will be used to determine whether the integrated dose exceeds the threshold values of > 1000 mrem TEDE or > 5000 mrem thyroid CDE at or beyond the site boundary. Actual meteorology is specifically identified since it gives the most accurate dose assessment. If the dose assessment results are available at the time that the classification is made, the results will be used in conjunction with this EAL for classifying the event rather than effluent radiation monitor EAL threshold values.

PNPS Basis Reference(s):

1. EP-IP-300, *"Offsite Radiological Dose Assessment"*

General Emergency - AG1.3

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

AG1.3 General Emergency

Field survey results indicate closed window dose rates > 1,000 mR/hr expected to continue for ≥ 1 hr at or beyond the site boundary

OR

Analyses of field survey samples indicate thyroid CDE > 5,000 mrem for 1 hr of inhalation at or beyond the site boundary

Mode Applicability:

All

NEI 99-01 Basis:

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

General Emergency - AG1.3

PNPS Basis:

Although this EAL initiating condition references the TEDE or thyroid CDE as an integrated dose, field survey results are not generally reported in these dose quantities, but rather in terms of a dose rate or an air sample concentration, respectively. For this reason, the field survey EALs are based on a closed window dose rate greater than 1000 mR/hr expected to continue for 1 hour or longer; or analyses of field survey samples indicate thyroid CDE greater than 5000 mrem for 1 hour of inhalation at or beyond the site boundary and whichever is more limiting.

PNPS Basis Reference(s):

1. EP-IP-310, *"Radiation Monitoring Team Activation And Response"*

Unusual Event - AU2.1

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 2 - Onsite Rad Conditions & Spent Fuel Pool Events

Initiating Condition: Unplanned rise in plant radiation levels

EAL:

AU2.1 Unusual Event

Unplanned low water level or alarm indicating uncontrolled water level decrease in the reactor cavity or spent fuel pool with **all** irradiated fuel assemblies remaining covered by water

AND

Valid area radiation monitor reading rise on ANY of the following:

- New Fuel Vault (RIS-1815-3D)
- Refuel Floor Shield Plug Area (RIS-1815-3E)
- Spent Fuel Pool Area (RIS-1815-3F)

Mode Applicability:

All

Unusual Event - AU2.1

NEI 99-01 Basis:

This EAL addresses increased radiation levels as a result of water level decreases above irradiated fuel or events that have resulted in, or may result in, unplanned increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant.

The refueling pathway is the reactor cavity connected to the spent fuel pool through the spent fuel pool gate. 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 refueling events where the water level drops below the RPV flange, classification would be via EAL CU2.1, CU2.2, or CU2.3. This event escalates to an Alert in accordance with EAL AA2.1 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 Table for events in operating modes 1-3.

PNPS Basis:

Loss of inventory from the reactor cavity or spent fuel pool may reduce water shielding above spent fuel and cause unexpected increases in plant radiation levels. Classification as an Unusual Event is warranted as a precursor to a more serious event.

Spent fuel pool level instruments LI-4816A and LI-4816B, located on the back wall of the Control Room, and Spent fuel pool temperature instruments TE-4831 and TS-4807, provide a wide range indication of level and temperature in the spent fuel pool. In addition, one or more of the following annunciators may be indicative of an uncontrolled water level decrease in the reactor cavity, spent fuel pool, or fuel transfer canal (cattle chute):

- **"SPENT FUEL POOL LEVEL LO"** (C903R-B2)
- **"FUEL POOL COOLING PANEL ALARM"** (C2R-D7)
- **"FUEL POOL LOW LEVEL"** (C39-F1)
- **"REACTOR BASIN LOW LEVEL"** (C39-E3)
- **"DRYWELL BELLOWS SEAL LEAKAGE"** (C39-A1)

Unusual Event - AU2.1

- **"REFUEL BELLOWS SEAL RUPTURE"** (C39-A2)
- **"FUEL POOL GATE LEAKAGE HIGH"** (C39-A3)
- **"SKIMMER SURGE TANK LOW LEVEL"** (C39-B2)
- **"FUEL POOL TEMP HIGH"** (C39-C1)

Spent Fuel Pool Skimmer Surge Tank level is indicated on LI-4815A (ref.1).

Increases in area radiation levels due to loss of water shielding above irradiated fuel outside the RPV may be detected by the following (ref. 2):

- Refuel Floor Area Radiation Monitor(s) in alarm.
- Refuel Floor Ventilation Process Radiation Monitor alarm.
- Portable radiation monitors

A minimum depth of water of about 10 feet over spent fuel assemblies and structures is ensured by providing no drain in the pool. In accordance with Technical Specifications Section 4.3.2, Drainage, the Spent Fuel Pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 115 ft (Tech Spec Amendment 177). (ref. 3)

This event escalates to an Alert if irradiated fuel outside the RPV is uncovered.

PNPS Basis Reference(s):

1. PNPS 2.4.31, *"Reactor Basin and/or Spent Fuel Pool Draindown"*
2. PNPS 5.4.3, *"Refueling Floor High Radiation"*
3. PNPS 2.2.85, *"Fuel Pool Cooling and Filtering System"*
4. EC 45088, Fukushima - Spent Fuel Pool Level Instrumentation

Unusual Event - AU2.2

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 2 - Onsite Rad Conditions & Spent Fuel Pool Events

Initiating Condition: Unplanned rise in plant radiation levels

EAL:

AU2.2 Unusual Event

Unplanned valid area radiation monitor reading or survey results rise by a factor of 1000 over normal levels*

* Normal levels can be considered as the highest reading in the past 24 hours excluding the current peak value

Mode Applicability:

All

NEI 99-01 Basis:

This EAL addresses increased radiation levels as a result of water level decreases above irradiated fuel or events that have resulted in, or may result in, unplanned increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant.

<u>EAL #2</u>

This EAL addresses increases in plant radiation levels that represent a loss of control of radioactive material resulting in a potential degradation in the level of safety of the plant.

This EAL excludes radiation level increases that result from planned activities such as use of radiographic sources and movement of radioactive waste materials. A specific list of ARMs is not required as it would restrict the applicability of the threshold. The intent is to identify loss of control of radioactive material in any monitored area.

PNPS Basis:

The ARMs monitor the gamma radiation levels in units of mR/hr at selected areas throughout the station. If radiation levels exceed a preset limit in any channel, the Main Control Room annunciator and local alarms will be energized to warn of abnormal or significantly changing radiological conditions (ref. 1, 2).

Routine and work-specific surveys are conducted throughout the station at frequencies specified by Radiation Protection management. Work-specific surveys are conducted in accordance with the Radiological Work Permit (RWP). (ref. 3)

PNPS Basis Reference(s):

1. PNPS 2.2.62, *"Area Radiation Monitoring System"*
2. PNPS 6.5-160, *"Calibration of the Area Radiation Monitoring System"*
3. PNPS 6.3-064, *"Routine Radiological Surveillance Program"*

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 2 - Onsite Rad Conditions & Spent Fuel Pool Events

Initiating Condition: Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the RPV

EAL:

AA2.1 Alert

Damage to irradiated fuel **OR** loss of water level (uncovering irradiated fuel outside the RPV) that causes a valid high alarm on **any** of the following radiation monitors (Panel C910/C911):

- New Fuel Vault (RIS-1815-3D)
- Refuel Floor Shield Plug Area (RIS-1815-3E)
- Spent Fuel Pool Area (RIS-1815-3F)
- Refuel Floor Vent Exhaust (RIS-1705-8A-D)

Mode Applicability:

All

NEI 99-01 Basis:

This EAL addresses 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 an actual or substantial potential degradation in the level of safety of the plant.

This EAL addresses radiation monitor indications of fuel uncovery and/or fuel damage.

Increased ventilation monitor readings may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background at the ventilation 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.

Escalation of this emergency classification level, if appropriate, would be based on EAL AS1.1, AS1.2, AS1.3, AG1.1, AG1.2, or AG1.3.

PNPS Basis:

When considering classification, information may come from:

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

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

Spent fuel pool level instruments LI-4816A and LI-4816B, located on the back wall of the Control Room, and Spent fuel pool temperature instruments TE-4831 and TS-4807, provide a wide range indication of level and temperature in the spent fuel pool. In addition, one or more of the following annunciators may be indicative of an uncontrolled water level decrease in the reactor cavity or spent fuel pool:

- **"SPENT FUEL POOL LEVEL LO"** (C903R-B2)
- **"FUEL POOL COOLING PANEL ALARM"** (C2R-D7)
- **"FUEL POOL LOW LEVEL"** (C39-F1)
- **"REACTOR BASIN LOW LEVEL"** (C39-E3)

- **"DRYWELL BELLOWS SEAL LEAKAGE"** C39-A1)
- **"REFUEL BELLOWS SEAL RUPTURE"** (C39-A2)
- **"FUEL POOL GATE LEAKAGE HIGH"** (C39-A3)
- **"SKIMMER SURGE TANK LOW LEVEL"** (C39-B2)
- **"FUEL POOL TEMP HIGH"** (C39-C1)

Spent Fuel Pool Skimmer Surge Tank level is indicated on LI-4815A (ref. 1).

The high alarms of the listed ARMs (RIS-1815-3D, RIS-1815-3E, and RIS-1815-3F) activate Control Room annunciator C7 (**"REFUEL FLOOR RAD HI"**) on Panel C904LC. The high alarm setpoint is at approximately 40 mR/hr. The ARM indication is displayed on recorder 40-RR-1815-6 (Panel C902) (ref. 2, 3).

The high alarm of the Refuel Floor Exhaust PRMs activates Control Room annunciators A4 (**"REFL FLR VENT RAD CHAN A HI"**) and B4 (**"REFL FLR VENT RAD CHAN B HI"**) on Panel C904LC. The high alarm is set at 16 mR/hr (increasing) during normal operation or ≤ 67 mR/hr (increasing) during refueling. The PRM indication is displayed on recorder RR-1705-21 on Panel C902. If both channels tripped, the following occurs (ref. 4, 5):

- Reactor Building ventilation system isolates
- Standby Gas Treatment System starts

Evacuation of refuel floor personnel to RB 91' changeout area is required if the high alarm is received on any of these monitors (ref. 6).

PNPS Basis Reference(s):

1. PNPS 2.4.31, *"Reactor Basin and/or Spent Fuel Pool Draindown"*
2. PNPS 6.5-160, *"Calibration of the Area Radiation Monitoring System"*
3. PNPS ARP-C904LC-C7
4. PNPS ARP-C904LC-A4
5. PNPS ARP-C904LC-B4
6. PNPS 5.4.3, *"Refueling Floor High Radiation"*
7. EC 45088, Fukushima - Spent Fuel Pool Level Instrumention

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 2 - Onsite Rad Conditions & Spent Fuel Pool Events

Initiating Condition: Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the RPV

EAL:

AA2.2 Alert

A water level drop in the reactor cavity or spent fuel pool that will result in irradiated fuel becoming uncovered

Mode Applicability:

All

NEI 99-01 Basis:

This EAL addresses 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 an actual or substantial potential degradation in the level of safety of the plant.

Escalation of this emergency classification level, if appropriate, would be based on EAL AS1.1, AS1.2, AS1.3, AG1.1, AG1.2, or AG1.3.

Alert - AA2.2

PNPS Basis:

When considering classification, information may come from:

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

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

Spent fuel pool level instruments LI-4816A and LI-4816B, located on the back wall of the Control Room, and Spent fuel pool temperature instruments TE-4831 and TS-4807, provide a wide range indication of level and temperature in the spent fuel pool. In addition, one or more of the following annunciators may be indicative of an uncontrolled water level decrease in the reactor cavity or spent fuel pool:

- **"SPENT FUEL POOL LEVEL LO"** (C903R-B2)
- **"FUEL POOL COOLING PANEL ALARM"** (C2R-D7)
- **"FUEL POOL LOW LEVEL"** (C39-F1)
- **"REACTOR BASIN LOW LEVEL"** (C39-E3)
- **"DRYWELL BELLOWS SEAL LEAKAGE"** (C39-A1)
- **"REFUEL BELLOWS SEAL RUPTURE"** (C39-A2)
- **"FUEL POOL GATE LEAKAGE HIGH"** (C39-A3)
- **"SKIMMER SURGE TANK LOW LEVEL"** (C39-B2)
- **"FUEL POOL TEMP HIGH"** (C39-C1)

Spent Fuel Pool Skimmer Surge Tank level is indicated on LI-4815A (ref. 1).

PNPS Basis Reference(s):

1. PNPS 2.4.31, *"Reactor Basin and/or Spent Fuel Pool Draindown"*
2. EC 45088, Fukushima - Spent Fuel Pool Level Instrumentation

Alert - AA3.1

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 3 - MCR/CAS Radiation

Initiating Condition: Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions

EAL:

AA3.1 Alert

Dose rates > 15 mR/hr in **EITHER** of the following areas requiring continuous occupancy to maintain plant safety functions:

Main Control Room (RIS-1815-2A, Panel C911)

OR

CAS (by survey)

Mode Applicability:

All

NEI 99-01 Basis:

This EAL addresses increased radiation levels that impact continued operation in areas requiring continuous occupancy to maintain safe operation or to perform a safe shutdown.

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

Areas requiring continuous occupancy include the Control Room and, as appropriate to the site, any other control stations that are staffed continuously, or a security alarm station.

PNPS Basis:

Areas that meet this threshold include the Main Control Room and the Central Alarm Station (CAS).

The Main Control Room Area Radiation Monitor (ARM) RIS-1815-2A on Panel C911 provides indication of area radiation levels in the Main Control Room. The high alarm of the Main Control Room ARM activates Control Room annunciator B7 (**"CONTROL ROOM RAD HI"**) on Panel C904LC. The high alarm is set at approximately 1 mR/hr. (ref. 1, 2, 3)

The CAS area has no permanently installed area radiation monitors that may be used to assess this EAL threshold. Therefore this threshold must be assessed via local radiation survey (ref. 4).

PNPS Basis Reference(s):

1. PNPS 2.2.62, *"Area Radiation Monitoring System"*
2. PNPS 6.5-160, *"Calibration of the Area Radiation Monitoring System"*
3. PNPS ARP-C904LC-B7
4. PNPS 6.3-064, *"Routine Radiological Surveillance Program"*

Category H - Hazards

EAL Group: ANY (EALs in this category are applicable to any plant condition, hot or cold.)

Hazards are non-plant, system-related events that can directly or indirectly affect plant operation, reactor plant safety, or personnel safety.

The events of this category pertain to the following subcategories:

1. Natural or Destructive Phenomena

Natural events include hurricanes, earthquakes, or tornados that have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety. Non-naturally occurring events that can cause damage to plant facilities and include aircraft crashes, missile impacts, etc.

2. Fire or Explosion

Fires can pose significant hazards to personnel and reactor safety. Appropriate for classification are fires within the site Protected Area or which may affect operability of vital equipment.

3. Hazardous Gas

Non-naturally occurring events that can cause damage to plant facilities and include toxic, corrosive, asphyxiant, or flammable gas leaks.

4. Security

Unauthorized entry attempts into the Protected Area, bomb threats, sabotage attempts, and actual security compromises threatening loss of physical control of the plant.

5. Control Room Evacuation

Events that are indicative of loss of Control Room habitability. If the Control Room must be evacuated, additional support for monitoring and controlling plant functions is necessary through the emergency response facilities.

6. Judgment

The EALs defined in other categories specify the predetermined symptoms or events that are indicative of emergency or potential emergency conditions and thus warrant classification. While these EALs have been developed to address the full spectrum of possible emergency conditions which may warrant classification and subsequent implementation of the Emergency Plan, a provision for classification of emergencies based on operator/management experience and judgment is still necessary. The EALs of this category provide the Emergency Director the latitude to classify emergency conditions consistent with the established classification criteria based upon Emergency Director judgment.

Unusual Event - HU1.1

Category: H - Hazards

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU1.1 Unusual Event

Seismic event identified by **any two** of the following:

- **"SEISMIC RECORDER OPERATING"** (C903R-B1) alarm
- Earthquake felt in plant
- National Earthquake Information Center

Mode Applicability:

All

NEI 99-01 Basis:

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

Damage may be caused to some portions of the site but should not affect ability of safety functions to operate.

A"felt earthquake" is an earthquake of sufficient intensity such that 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.

The National Earthquake Center can confirm whether an earthquake has occurred in the area of the plant.

Unusual Event - HU1.1

PNPS Basis:

The method of detection with respect to emergency classification relies on the agreement of the shift operators on duty in the Control Room that the suspected ground motion is a felt earthquake as well as the actuation of the PNPS seismic instrumentation. Consensus of the Control Room operators with respect to ground motion helps avoid unnecessary classification if the seismic switches inadvertently trip or detect vibrations not related to an earthquake.

Ground motion acceleration of 0.01g activates Control Room annunciator **"SEISMIC RECORDER OPERATING"** (C903R-B1) (ref. 1).

The National Earthquake Center can confirm whether an earthquake has occurred in the area of the plant.

The SYSCOM seismic monitoring system is actuated by the trigger function of the recorder/sensors, which are located in the Reactor Building El. -17' CRD Quadrant, Reactor Building El. 23', and Reactor Building El. 91'. The trigger setpoint is 0.01g and the logic to actuate the Control Room annunciator is any one of the three recorder/sensors. The trigger actuation in any recorder/sensor will initiate the recording function in the other recorder/sensors and actuate Control Room indication for Operators in Panel C911. There are three indications to alert Operators that the trigger was actuated and that the seismic recording was/is activated (ref. 1).

- The indicator panel that is part of ECU-370 has three red LEDs designated for the trigger function. There is one for each recorder/sensor; at least one of these is illuminated.
- The red trigger LED for ECU-370 is illuminated indicating the logic for the system has been met.
- Control Room annunciator **"SEISMIC RECORDER OPERATING"** (C903R-B1) will be in alarm whenever the trigger senses a magnitude equal to or greater than 0.01g or 10mg.

This event escalates to an Alert under EAL HA1.1 if the earthquake exceeds Operating Basis Earthquake (OBE) levels.

PNPS Basis Reference(s):

1. PNPS 5.2.1, *"Earthquake"*

Unusual Event - HU1.2

Category: H - Hazards

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU1.2 Unusual Event

Tornado striking within Protected Area boundary

OR

Sustained high winds > 105 mph

Mode Applicability:

All

NEI 99-01 Basis:

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL is based on a tornado striking (touching down) or high winds within the Protected Area.

Escalation of this emergency classification level, if appropriate, would be based on visible damage or by other in plant conditions via EAL HA1.2.

Unusual Event - HU1.2

PNPS Basis:

A tornado striking (touching down) within the Protected Area warrants declaration of an Unusual Event regardless of the measured wind speed at the meteorological tower. A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

Design wind speed of 105 mph is calculated from the design basis wind loadings (ref. 1). The range of the Climatronics Model #100075 F460 Wind Speed Sensor is 0.5 to 145 mph (ref. 2).

PNPS Basis Reference(s):

1. SUDDSRF96-35 Wind and Tornado Evaluation for PNPS
2. Modification ER03114340

Unusual Event - HU1.3

Category: H - Hazards

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the Protected Area (turbine)

EAL:

HU1.3 Unusual Event

Turbine failure resulting in casing penetration or damage to turbine or generator seals

Mode Applicability:

All

NEI 99-01 Basis:

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. 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.

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 EAL HU2.1 and EAL HU3.1.

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 this emergency classification level, if appropriate, would be to EAL HA1.4 based on damage done by projectiles generated by the failure or by the radiological releases. These latter events would be classified by the EALs in Category A or Category F.

Unusual Event - HU1.3

PNPS Basis:

The turbine generator stores large amounts of rotational kinetic energy in its rotor. In the unlikely event of a major mechanical failure, this energy may be transformed into both rotational and translational energy of rotor fragments. These fragments may impact the surrounding stationary parts. If the energy-absorbing capability of these stationary turbine generator parts is insufficient, external missiles will be released. These ejected missiles may impact various plant structures, including those housing safety related equipment.

In the event of missile ejection, the probability of a strike on a plant region is a function of the energy and direction of an ejected missile and of the orientation of the turbine with respect to the plant region.

PNPS Basis Reference(s):

None

Unusual Event - HU1.4

Category: H - Hazards

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU1.4 Unusual Event

Flooding in **any** Table H-2 area that has the potential to affect safety-related equipment needed for the current operating mode

Table H-2 Internal Flooding Areas
- Reactor Building
- Rx Closed Cooling Water System Auxiliary Bays
- Turbine Building
- Diesel Generator Rooms
- Salt Service Water Bays
- Radwaste Area

Mode Applicability:

All

Unusual Event - HU1.4

NEI 99-01 Basis:

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps.

Escalation of this emergency classification level, if appropriate, would be based on visible damage via EAL HA1.5 or by other plant conditions.

PNPS Basis:

The internal flooding areas of concern are listed in Table H-2 Internal Flooding Areas.

Flooding in these areas could have the potential to cause a reactor trip and could result in consequential failures to important systems. The potential for flooding in these areas was determined by an examination of piping systems in the area and also considered propagation of water from one area to another (ref. 1).

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

PNPS Basis Reference(s):

1. PNPS-NE-07-00006 Rev. 0

Unusual Event - HU1.5

Category: H - Hazards

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU1.5 Unusual Event

Seawater bay water level > +13'6" MSL (LI-3831A/B)

OR

Seawater bay water level < -13'9" MSL (LI-3831A/B)

Mode Applicability:

All

NEI 99-01 Basis:

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses other site-specific phenomena (such as flood) that can also be precursors of more serious events.

PNPS Basis:

As illustrated in Figure H-1 (ref. 1, 2), ground level at the screenhouse is +21'6" MSL and well below the flood level of +13'6" MSL. Since the entrances to all structures containing equipment necessary for reactor shutdown and cooling are at elevations well above +13'6" MSL, they are protected against flooding from external sources. Seawater bay water level < -13'9" MSL is the design minimum level for the SSW pumps.

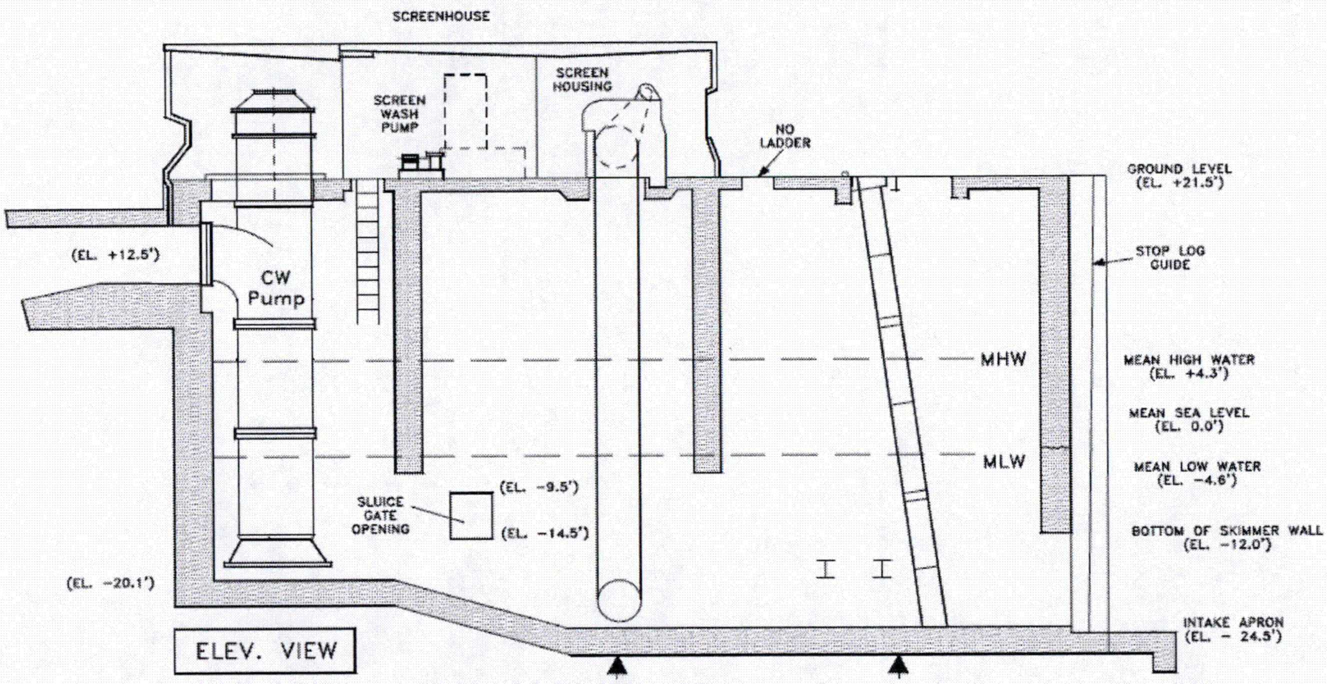
Unusual Event - HU1.5

PNPS Basis Reference(s):

1. PNPS 2.2.94, *"Seawater System"*
2. PNPS 2.4.154, *"Intake Structure Fouling"*
3. PNPS 2.2.32, *"Salt Service Water System (SSW)"*
4. PNPS 8.E.29, *"Salt Service Water System Instrumentation Calibration"*

Unusual Event - HU1.5

Figure H-1 Screenhouse Bay Water Levels



+16'0"	Maximum Monitored Water Level
+13'6"	Flood Level
+4'13"	Mean High Tide
0'0"	Mean Sea Level
-4'8"	Mean Low Tide
-7'0"	Design Low Water Level
-10'0"	PNPS 2.4.154 action level to reduce Reactor power and secure affected Seawater Pump
-13'9"	Design Minimum Level for SSW Pumps
-14'6"	Bottom of sluice gates between seawater bays and SSW bays
-15'0"	Calculated worst case level. PNPS 2.4.154 action level to secure affected Seawater Pump (8' below design low water level).
-16'0"	Minimum Monitored Water Level

Alert - HA1.1

Category: H - Hazards

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the plant Vital Area

EAL:

HA1.1 Alert

Seismic event identified by **any two** of the following:

- **"SEISMIC RECORDER OPERATING"** (C903R-B1) alarm
- Earthquake felt in plant
- National Earthquake Information Center

AND EITHER:

Ground motion > Operating Basis Earthquake (0.08g) per analysis

OR

Control Room indication of degraded performance of systems required for the safe shutdown of the plant

Mode Applicability:

All

Alert - HA1.1

NEI 99-01 Basis:

This EAL escalates from HU1.1 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 of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

Seismic events of this magnitude can result in a Vital Area being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

The National Earthquake Information Center can confirm whether an earthquake has occurred in the area of the plant.

PNPS Basis:

Ground motion acceleration of 0.08g horizontal or vertical is the Operating Basis Earthquake for PNPS (ref. 1).

Ground motion acceleration of 0.01g activates Control Room annunciator **"SEISMIC RECORDER OPERATING"** (C903R-B1) (ref. 1).

The SYSCOM seismic monitoring system is actuated by the trigger function of the recorder/sensors, which are located in the Reactor Building El. -17' CRD Quadrant, Reactor Building El. 23', and Reactor Building El. 91'. The trigger setpoint is 0.01g and the logic to actuate the Control Room annunciator is any one of the three recorder/sensors. The trigger actuation in any recorder/sensor will initiate the recording function in the other recorder/sensors and actuate Control Room indication for Operators in Panel C911. There are three indications to alert Operators that the trigger was actuated and that the seismic recording was/is activated (ref. 1).

- The indicator panel that is part of ECU-370 has three red LEDs designated for the trigger function. There is one for each recorder/sensor; at least one of these is illuminated.
- The red trigger LED for ECU-370 is illuminated indicating the logic for the system has been met.
- Control Room annunciator **"SEISMIC RECORDER OPERATING"** (C903R-B1) will be in alarm whenever the trigger senses a magnitude equal to or greater than 0.01g or 10mg.

PNPS 5.2.1, *"Earthquake,"* provides the guidance for determining whether the OBE earthquake threshold is exceeded by analysis.

PNPS Basis Reference(s):

1. PNPS 5.2.1, *"Earthquake"*

Alert - HA1.2

Category: H - Hazards

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the plant Vital Area

EAL:

HA1.2 Alert

Tornado or sustained high winds > 105 mph within Protected Area boundary resulting in visible damage to **any** Table H-1 plant structures/equipment or Control Room indication of degraded performance of safety systems

Table H-1 Safe Shutdown Areas
- Reactor Building
- Control Room
- Cable Spreading Room
- 4160 Switchgear Rooms
- Diesel Generator Rooms
- Salt Service Water Bays
- Rx Closed Cooling Water System Auxiliary Bays
- Standby Gas Treatment Area, 51' Level, Turbine Bldg
- Condensate Storage Tank

Mode Applicability:

All

Alert - HA1.2

NEI 99-01 Basis:

This EAL escalates from HU1.2 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 of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL is based on a tornado striking (touching down) or high winds that have caused visible damage to structures containing functions or systems required for safe shutdown of the plant.

PNPS Basis:

This threshold addresses events that may have resulted in Safe Shutdown Areas being subjected to forces (tornado or high winds > 105 mph, ref. 1) beyond design limits and thus damage may be assumed to have occurred to plant safety systems. Table H-1 Safe Shutdown Areas house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained shutdown (ref. 2, 3, 4).

A tornado striking (touching down) within the Protected Area resulting in visible damage warrants declaration of an Alert regardless of the measured wind speed at the meteorological tower. A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

PNPS Basis Reference(s):

1. SUDDSRF96-35, Wind and Tornado Evaluation for PNPS
2. PNPS 5.5.1, *"General Fire Procedure"*
3. PNPS 5.5.2, *"Special Fire Procedure"*
4. PNPS 1.3.40, *"Specifications For Vital Area Barrier Openings, Degradation, And Repair"*

Alert - HA1.3

Category: H - Hazards

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the plant Vital Area

EAL:

HA1.3 Alert

Vehicle crash resulting in visible damage to **any** Table H-1 plant structures or equipment or Control Room indication of degraded performance of safety systems

Table H-1 Safe Shutdown Areas
- Reactor Building
- Control Room
- Cable Spreading Room
- 4160 Switchgear Rooms
- Diesel Generator Rooms
- Salt Service Water Bays
- Rx Closed Cooling Water System Auxiliary Bays
- Standby Gas Treatment Area, 51' Level, Turbine Bldg
- Condensate Storage Tank

Mode Applicability:

All

Alert - HA1.3

NEI 99-01 Basis:

This EAL escalates from an Unusual Event 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 of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL addresses vehicle crashes within the Protected Area that result in visible damage to vital areas or indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant.

PNPS Basis:

Table H-1 Safe Shutdown Areas house equipment the operation of which may be needed to ensure the reactor reaches and is maintained in shutdown (ref. 1, 2, 3).

If the vehicle crash is determined to be hostile in nature, the event is classified under EAL HS4.1.

PNPS Basis Reference(s):

1. PNPS 5.5.1, *"General Fire Procedure"*
2. PNPS 5.5.2, *"Special Fire Procedure"*
3. PNPS 1.3.40, *"Specifications For Vital Area Barrier Openings, Degradation, And Repair"*

Alert - HA1.4

Category: H - Hazards

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the plant Vital Area

EAL:

HA1.4 Alert

Turbine failure-generated missiles result in **any** visible damage to or penetration of **any** Table H-1 area

Table H-1 Safe Shutdown Areas
- Reactor Building
- Control Room
- Cable Spreading Room
- 4160 Switchgear Rooms
- Diesel Generator Rooms
- Salt Service Water Bays
- Rx Closed Cooling Water System Auxiliary Bays
- Standby Gas Treatment Area, 51' Level, Turbine Bldg
- Condensate Storage Tank

Mode Applicability:

All

Alert - HA1.4

NEI 99-01 Basis:

This EAL escalates from HU1.3 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 of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL addresses the threat to safety related equipment imposed by projectiles generated by main turbine rotating component failures. Therefore, this EAL is consistent with the definition of an Alert in that the potential exists for actual or substantial potential degradation of the level of safety of the plant.

PNPS Basis:

The turbine generator stores large amounts of rotational kinetic energy in its rotor. In the unlikely event of a major mechanical failure, this energy may be transformed into both rotational and translational energy of rotor fragments. These fragments may impact the surrounding stationary parts. If the energy-absorbing capability of these stationary turbine generator parts is insufficient, external missiles will be released. These ejected missiles may impact various plant structures, including those housing safety related equipment.

In the event of missile ejection, the probability of a strike on a plant region is a function of the energy and direction of an ejected missile and of the orientation of the turbine with respect to the plant region.

Table H-1 Safe Shutdown Areas house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained shutdown (ref. 1, 2, 3).

Alert - HA1.4

PNPS Basis Reference(s):

1. PNPS 5.5.1, *"General Fire Procedure"*
2. PNPS 5.5.2, *"Special Fire Procedure"*
3. PNPS 1.3.40, *"Specifications for Vital Area Barrier Openings, Degradation, and Repair"*

Alert - HA1.5

Category: H - Hazards

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the plant Vital Area

EAL:

HA1.5 Alert

Flooding in **any** Table H-2 area 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

Table H-2 Internal Flooding Areas
- Reactor Building
- Rx Closed Cooling Water System Auxiliary Bays
- Turbine Building
- Diesel Generator Rooms
- Salt Service Water Bays
- Radwaste Area

Mode Applicability:

All

NEI 99-01 Basis:

This EAL escalates from EAL HU1.4 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 of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate, or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant.

Flooding, as used in this EAL, describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

PNPS Basis:

The internal flooding areas of concern are listed in Table H-2 Internal Flooding Areas.

Flooding in these areas could have the potential to cause a reactor trip and could result in consequential failures to important systems. The potential for flooding in these areas was determined by an examination of piping systems in the areas and also considered propagation of water from one area to another (ref. 1).

Alert - HA1.5

PNPS Basis Reference(s):

1. PNPS-NE-07-00006 Rev. 0

Category: H - Hazards

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the plant Vital Area

EAL:

HA1.6 Alert

Seawater bay water level > +16'0" MSL (LI-3831A/B)

OR

Seawater bay water level < -16'0" MSL(LI-3831A/B)

Mode Applicability:

All

NEI 99-01 Basis:

This EAL escalates from HU1.5 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 of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL addresses other PNPS phenomena that result in visible damage to vital areas or results in indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant (such as flood) that can also be precursors of more serious events.

Alert - HA1.6

PNPS Basis:

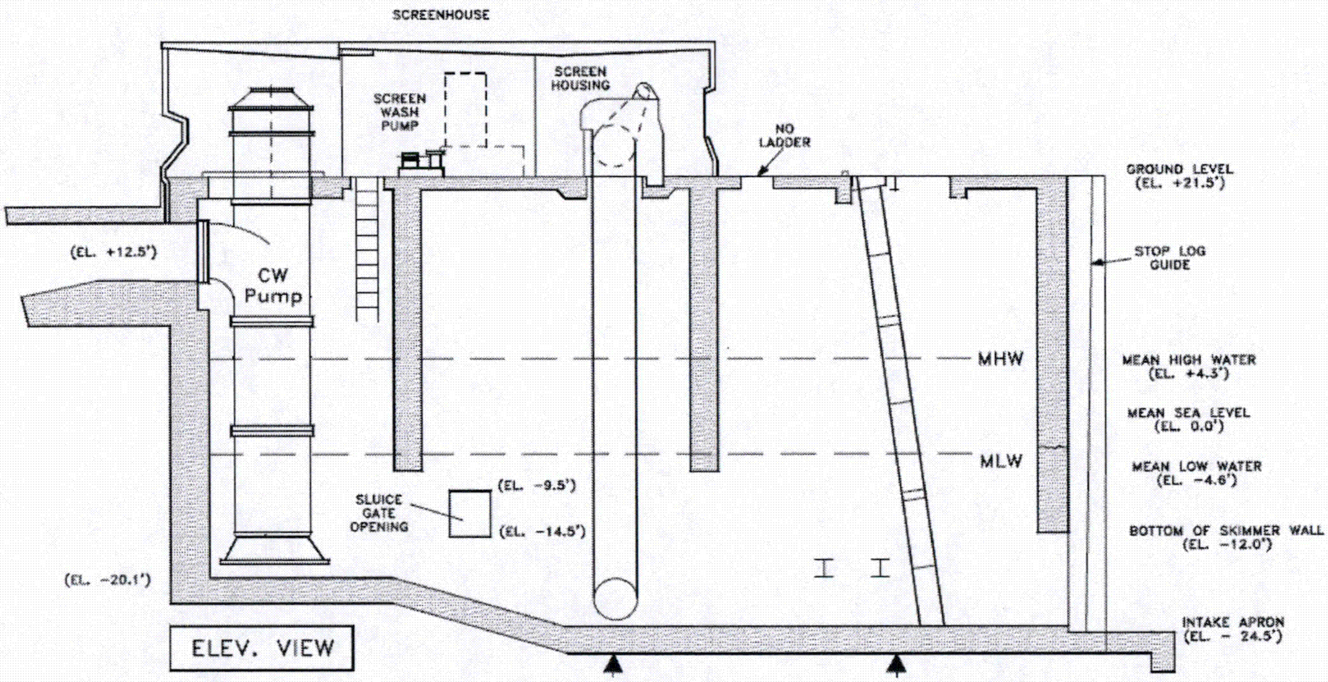
As illustrated in Figure H-1 (ref. 1, 2), ground level at the screenhouse is +21'6" MSL and well below the flood level of +13'6" MSL. Since the entrances to all structures containing equipment necessary for reactor shutdown and cooling are at elevations well above +16'0" MSL, they are protected against flooding from external sources. The specified water levels are the maximum and minimum monitored seawater bay water levels.

PNPS Basis Reference(s):

1. PNPS 2.2.94, *"Seawater System"*
2. PNPS 2.4.154, *"Intake Structure Fouling"*
3. PNPS 2.2.32, *"Salt Service Water System (SSW)"*
4. PNPS 8.E.29, *"Salt Service Water System Instrumentation Calibration"*

Alert - HA1.6

Figure H-1 Screenhouse Bay Water Levels



+16'0"	Maximum Monitored Water Level
+13'6"	Flood Level
+4'13"	Mean High Tide
0'0"	Mean Sea Level
-4'8"	Mean Low Tide
-7'0"	Design Low Water Level
-10'0"	PNPS 2.4.154 action level to reduce Reactor power and secure affected Seawater Pump
-13'9"	Design Minimum Level for SSW Pumps
-14'6"	Bottom of sluice gates between seawater bays and SSW bays
-15'0"	Calculated worst case level. PNPS 2.4.154 action level to secure affected Seawater Pump (8' below design low water level).
-16'0"	Minimum Monitored Water Level

Unusual Event - HU2.1

Category: H - Hazards

Subcategory: 2 - Fire or Explosion

Initiating Condition: Fire within the Protected Area not extinguished within 15 min. of detection or explosion within Protected Area boundary

EAL:

HU2.1 Unusual Event

Fire not extinguished within 15 min. of Control Room notification or verification of a Control Room fire alarm in **any** Table H-1 area (Note 3)

Note 3: The Emergency Director should not wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table H-1 Safe Shutdown Areas
- Reactor Building
- Control Room
- Cable Spreading Room
- 4160 Switchgear Rooms
- Diesel Generator Rooms
- Salt Service Water Bays
- Rx Closed Cooling Water System Auxiliary Bays
- Standby Gas Treatment Area, 51' Level, Turbine Bldg
- Condensate Storage Tank

Unusual Event - HU2.1

Mode Applicability:

All

NEI 99-01 Basis:

This EAL addresses the magnitude and extent of fires that may be potentially significant precursors of damage to safety systems. It addresses the fire and not the degradation in performance of affected systems that may result.

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 fire detection system alarm/actuation. Verification of a fire detection system alarm/actuation includes actions that can be taken within the Control Room or other nearby site-specific location to ensure that it is not spurious. An 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).

Escalation of this emergency classification level, if appropriate, would be based on EAL HA2.1.

PNPS Basis:

Fire, as used in this EAL, means 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.

Fires that are not within an H-1 area do not meet the criteria for an UE classification. If a fire alarm in an H-1 area cannot be disproved within the 15 minutes allowed, the UE needs to be performed.

Unusual Event - HU2.1

PNPS Basis Reference(s):

1. PNPS 5.5.1, *"General Fire Procedure"*
2. PNPS 5.5.2, *"Special Fire Procedure"*
3. PNPS 1.3.40, *"Specifications for Vital Area Barrier Openings, Degradation, and Repair"*

Unusual Event - HU2.2

Category: H - Hazards

Subcategory: 2 - Fire or Explosion

Initiating Condition: Fire within the Protected Area not extinguished within 15 min. of detection or explosion within Protected Area boundary

EAL:

HU2.2	**Unusual Event**
Explosion within the Protected Area	

Mode Applicability:

All

NEI 99-01 Basis:

This EAL addresses the magnitude and extent of explosions that may be potentially significant precursors of damage to safety systems. It addresses the explosion and not the degradation in performance of affected systems that may result.

As used here, detection is visual observation and report by plant personnel or sensor alarm indication.

This EAL addresses only those explosions of sufficient force to damage permanent structures or equipment within the Protected Area.

No attempt is made to assess the actual magnitude of the damage. The occurrence of the explosion is sufficient for declaration.

The Emergency Director also needs to consider any security aspects of the explosion, if applicable.

Escalation of this emergency classification level, if appropriate, would be based on EAL HA2.1.

Unusual Event - HU2.2

PNPS Basis:

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

A steam line break or steam explosion that damages surrounding permanent structures or equipment would be classified under this EAL. This does not mean the emergency is classified simply because the steam line break occurred. The method of damage is not as important as the degradation of plant structures or equipment. The need to classify the steam line break itself is considered in fission product barrier degradation monitoring (EALs in Category F).

If the explosion is determined to be hostile in nature, the event is classified under security based EALs.

PNPS Basis Reference(s):

None

Alert - HA2.1

Category: H - Hazards

Subcategory: 2 - Fire or Explosion

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

EAL:

HA2.1 Alert

Fire or explosion resulting in visible damage to **any** Table H-1 area containing safety systems or components or Control Room indication of degraded performance of safety systems

Table H-1 Safe Shutdown Areas
- Reactor Building
- Control Room
- Cable Spreading Room
- 4160 Switchgear Rooms
- Diesel Generator Rooms
- Salt Service Water Bays
- Rx Closed Cooling Water System Auxiliary Bays
- Standby Gas Treatment Area, 51' Level, Turbine Bldg
- Condensate Storage Tank

Mode Applicability:

All

Alert - HA2.1

NEI 99-01 Basis:

Visible damage is used to identify the magnitude of the fire or explosion and to discriminate against minor fires and explosions.

The reference to structures containing safety systems or components is included to discriminate against fires or 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 or explosion was large enough to cause damage to these systems.

The use of visible damage should not be interpreted as mandating a lengthy damage assessment prior to classification. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency Director with the resources needed to perform detailed damage assessments.

The Emergency Director also needs to consider any security aspects of the explosion.

Escalation of this emergency classification level, if appropriate, will be based on system malfunctions, fission product barrier degradation, or abnormal rad levels/radiological effluent EALs.

PNPS Basis:

Fire, as used in this EAL, means 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.

An explosion is a rapid, violent, unconfined combustion or a catastrophic failure of pressurized equipment that potentially imparts significant energy to nearby structures and materials.

A steam line break or steam explosion that damages permanent structures or equipment would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment. The need to classify the steam line break itself is considered in fission product barrier degradation monitoring (EAL Category F).

PNPS Basis Reference(s):

1. PNPS 5.5.1, *"General Fire Procedure"*
2. PNPS 5.5.2, *"Special Fire Procedure"*
3. PNPS 1.3.40, *"Specifications For Vital Area Barrier Openings, Degradation, And Repair"*

Unusual Event - HU3.1

Category: H - Hazards

Subcategory: 3 - Hazardous Gas

Initiating Condition: Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal plant operations

EAL:

HU3.1 Unusual Event

Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect normal plant operations

Mode Applicability:

All

NEI 99-01 Basis:

This EAL is based on the release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect normal plant operations.

The fact that SCBA may be worn does not eliminate the need to declare the event.

This IC is not intended to require significant assessment or quantification. It assumes an uncontrolled process that has the potential to affect plant operations. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness, or even death.

Escalation of this emergency classification level, if appropriate, would be based on EAL HA3.1.

Unusual Event - HU3.1

PNPS Basis:

As used in this EAL, affecting normal plant operations means that activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures, have been impacted. For example, the discharge of hydrogen gas from the main generator during plant shutdown activities would not constitute an emergency classification unless the release adversely affected normal plant operations. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from normal plant operations and thus would be considered to have been affected.

The release may have originated within the plant site areas or it may have originated offsite and subsequently drifted onto plant site areas. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) resulting in the plant being within the evacuation area should also be considered in this EAL because of the adverse affect on normal plant operations.

Some gases are toxic by their very nature. Others, like carbon dioxide, can be lethal if it reduces oxygen to low concentrations (asphyxiant) that are immediately dangerous to life and health (IDLH). Oxygen deficient atmospheres (less than 19.5% oxygen) are considered IDLH (ref. 1). NRC position is that any time carbon dioxide is discharged in plant areas such that the area becomes uninhabitable, regardless of whether anyone is in the areas, conditions for classification exist. The EAL assumes an uncontrolled process that has the potential to affect plant operations or personnel safety. Releases occurring during planned surveillance activities or planned maintenance/tagout activities, therefore, are excluded.

The following documents provide additional information on hazardous substances and spills.

- PNPS 5.5.4, *"Response to Hazardous Material Incidents"* (ref. 2)
- Regulatory Guide 1.78, Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, Table 1, Toxicity Limits (IDLH Limits) for Some Hazardous Chemicals

Should the release affect access to plant safe shutdown areas, escalation to an Alert would be based on EAL HA3.1. Should an explosion or fire occur due to flammable gas within an affected plant area, an Alert may be appropriate based on EAL HA2.1.

Unusual Event - HU3.1

PNPS Basis Reference(s):

1. PNPS 1.4.12, *"Primary Containment Entry"*
2. PNPS 5.5.4, *"Response to Hazardous Material Incidents"*

Unusual Event - HU3.2

Category: H - Hazards

Subcategory: 3 - Hazardous Gas

Initiating Condition: Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal operation of the plant

EAL:

HU3.2 Unusual Event

Recommendation by local, county, or state officials to evacuate or shelter site personnel based on offsite event

Mode Applicability:

All

NEI 99-01 Basis:

This EAL is based on the release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect normal plant operations.

The fact that SCBA may be worn does not eliminate the need to declare the event.

This IC is not intended to require significant assessment or quantification. It assumes an uncontrolled process that has the potential to affect plant operations. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness, or even death.

Escalation of this emergency classification level, if appropriate, would be based on EAL HA3.1.

Unusual Event - HU3.2

PNPS Basis:

This EAL is based on the existence of an uncontrolled release originating offsite and local, county, or state officials have reported the need for evacuation or sheltering of site personnel. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) are considered in this EAL because they may adversely affect normal plant operations.

State officials may determine the evacuation area for offsite spills by using the Department of Transportation (DOT) Evacuation Tables for Selected Hazardous Materials in the DOT Emergency Response Guide for Hazardous Materials.

Should the release affect plant safe shutdown areas, escalation to an Alert would be based on EAL HA3.1. Should an explosion or fire occur due to flammable gas within an affected plant area, an Alert may be appropriate based on EAL HA2.1.

PNPS Basis Reference(s):

None

Alert - HA3.1

Category: H - Hazards

Subcategory: 3 - Hazardous Gas

Initiating Condition: Access to a vital area is prohibited due to release of toxic, corrosive, asphyxiant or flammable gases which jeopardizes operation of systems required to maintain safe operations or safely shut down the reactor

EAL:

HA3.1 Alert

Access to **any** Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shut down the reactor (Note 8)

Note 8: If the equipment in the stated area was already inoperable or out of service before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event

Table H-1 Safe Shutdown Areas
- Reactor Building
- Control Room
- Cable Spreading Room
- 4160 Switchgear Rooms
- Diesel Generator Rooms
- Salt Service Water Bays
- Rx Closed Cooling Water System Auxiliary Bays
- Standby Gas Treatment Area, 51' Level, Turbine Bldg
- Condensate Storage Tank

Alert - HA3.1

Mode Applicability:

All

NEI 99-01 Basis:

Gases in a vital area can affect the ability to safely operate or safely shut down the reactor.

The fact that SCBA may be worn does not eliminate the need to declare the event.

Declaration should not be delayed for confirmation from atmospheric testing if the atmosphere poses an immediate threat to life and health or an immediate threat of severe exposure to gases. This could be based upon documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards.

If the equipment in the stated area was already inoperable or out of service before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shut down beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness, or even death.

An uncontrolled release of flammable gases 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. Flammable gases, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This EAL assumes concentrations of flammable gases which can ignite/support combustion.

Escalation of this emergency classification level, if appropriate, will be based on system malfunctions, fission product barrier degradation, or abnormal rad levels/radioactive effluent EALs.

Alert - HA3.1

PNPS Basis:

This EAL is based on gases that have entered a plant structure in concentrations that could be unsafe for plant personnel and, therefore, preclude access to equipment necessary for the safe operation of the plant. Table H-1 Safe Shutdown Areas contains systems that are operated to establish or maintain safe shutdown (ref. 1, 2, 3).

The fact that SCBA may be worn does not eliminate the need to declare the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness, or even death.

Flammable gases, 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 gases 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.

This EAL does not apply to routine inerting of the Primary Containment.

The following documents provide additional information on hazardous substances and spills.

- PNPS 5.5.4, *"Response to Hazardous Material Incidents"* (ref. 4)
- Regulatory Guide 1.78, Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, Table 1, Toxicity Limits (IDLH Limits) for Some Hazardous Chemicals

PNPS Basis Reference(s):

1. PNPS 5.5.1, *"General Fire Procedure"*
2. PNPS 5.5.2, *"Special Fire Procedure"*
3. PNPS 1.3.40, *"Specifications for Vital Area Barrier Openings, Degradation, and Repair"*
4. PNPS 5.5.4, *"Response to Hazardous Material Incidents"*

Category: H - Hazards

Subcategory: 4 - Security

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

EAL:

HU4.1 Unusual Event

A security condition that does **not** involve a hostile action as reported by the Station Security Force

OR

A credible site-specific security threat notification

OR

A validated notification from NRC providing information of an aircraft threat

Mode Applicability:

All

NEI 99-01 Basis:

Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10CFR73.71 or in some cases under 10CFR50.72. Security events assessed as hostile actions are classifiable under EALs HA4.1, HS4.1, and HG4.1.

A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification level in accordance with the site's Safeguards Contingency Plan and Emergency Plan.

Unusual Event - HU4.1

First Threshold

Reference is made to PNPS Security Force because these individuals are the designated personnel onsite 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 PNPS Safeguards Contingency Plan.

This threshold is based on PNPS security plans. PNPS Safeguards Contingency Plans are based on guidance provided by NEI 03-12.

Second Threshold

This threshold is included to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the Unusual Event.

The determination of "credible" is made through use of information found in the PNPS Safeguards Contingency Plan.

Third Threshold

The intent of this threshold is to ensure that notifications for the aircraft threat are made in a timely manner and that offsite response organizations (OROs) and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this threshold to replace existing nonhostile-related EALs involving aircraft.

This threshold is met when PNPS receives information regarding an aircraft threat from the NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Unusual Event.

The NRC Headquarters Operations Officer (HOO) will communicate to PNPS if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Escalation to the Alert emergency classification level via EAL HA4.1 would be appropriate if the threat involves an airliner within 30 minutes of the plant.

Unusual Event - HU4.1

PNPS Basis:

Security EALs could involve events due to nonhostile and hostile actions. It is important to understand the differences between these actions and therefore the definitions of hostile action and hostile force are provided with this basis.

Hostile Action: An act toward PNPS or its plant personnel that includes the use of violent force to destroy equipment, takes hostages, and/or intimidates plant personnel to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on PNPS. Nonterrorism-based EALs should be used to address such activities (e.g., violent acts between individuals in the Owner Controlled Area.)

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

FIRST THRESHOLD: The declaration of an Unusual Event for a significant security condition (e.g., an attempted unauthorized entry into the Protected Area) is appropriate to ensure positive action is taken to protect personnel safety and to notify offsite law enforcement agencies.

The term "... as reported by the Station Security Force" infers that Station Security has evaluated the incident and has determined it is a significant security condition. **These individuals are the designated onsite 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 controls placed on the PNPS Safeguards Contingency Plan.**

SECOND THRESHOLD: The declaration of an Unusual Event for any credible security threat is appropriate to ensure positive action is taken to maintain site security, protect personnel safety, and notify offsite law enforcement agencies. The term "credible security threat" refers to a threat which in the judgment of the Station Security Force is credible as specified in the PNPS Safeguards Contingency Plan.

THIRD THRESHOLD: PNPS 5.3.14.1, *"Airborne Threat"*, describes the actions to be taken when notified of an airborne threat.

Unusual Event - HU4.1

PNPS Basis Reference(s):

1. NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security Based Events", July 18, 2005
2. NEI White Paper, "Enhancements to Emergency Preparedness Programs for Hostile Action", November 18, 2005
3. PNPS 5.3.14, *"Security Incidents"*
4. PNPS 5.3.14.1, *"Airborne Threat"*

Category: H - Hazards

Subcategory: 4 - Security

Initiating Condition: Hostile action within the Owner Controlled Area or airborne attack threat

EAL:

HA4.1 Alert

A hostile action is occurring or has occurred within the **Owner Controlled Area** as reported by the Station Security Force

OR

A validated notification from NRC of an airliner attack threat within 30 min. of the site

Mode Applicability:

All

NEI 99-01 Basis:

Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land, or water attack elements.

The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as onsite evacuation, dispersal, or sheltering).

Alert - HA4.1

First Threshold

This threshold addresses the potential for a very rapid progression of events due to a hostile action. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the OCA. Those events are adequately addressed by other EALs.

Note that this threshold is applicable for any hostile action occurring, or that has occurred, in the Owner Controlled Area.

Second Threshold

This threshold addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time.

The intent of this threshold is to ensure that notifications for the airliner attack threat are made in a timely manner and that offsite response organizations (OROs) and plant personnel are at a state of heightened awareness regarding the credible threat. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.

This threshold is met when a plant receives information regarding an airliner attack threat from the NRC and the airliner is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Alert - HA4.1

PNPS Basis:

Security EALs could involve events due to nonhostile and hostile actions. It is important to understand the differences between these actions and therefore the definitions of hostile action and hostile force are provided with this basis.

Hostile Action: An act toward PNPS or its plant personnel that includes the use of violent force to destroy equipment, takes hostages, and/or intimidates plant personnel to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on PNPS. Nonterrorism-based EALs should be used to address such activities (e.g., violent acts between individuals in the Owner Controlled Area.)

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

<u>FIRST THRESHOLD</u>: This EAL is intended to address the potential for a very rapid progression of events due to an attack including:

- Air attack (aircraft impacting the Owner Controlled Area)
- Land-based attack (hostile force progressing across PNPS property or directing projectiles at the site)
- Waterborne attack (hostile force on water attempting forced entry or directing projectiles at the site)
- Bombs

This EAL is intended to address the contingency for a very rapid progression of events due to a hostile attack and the possibility for additional attack. This EAL is not premised solely on the potential for a radiological release. Rather, the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional attack elements. It is appropriate for offsite response organizations to be notified and to activate in order to be better prepared to respond should protective actions become necessary.

Alert - HA4.1

For the purpose of this EAL, "Owner Controlled Area (OCA)" is the Entergy owned property on the north side of Rocky Hill Road.

SECOND THRESHOLD: PNPS 5.3.14.1, *"Airborne Threat"*, describes the action to be taken when notified of an airborne threat.

PNPS Basis Reference(s):

1. NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security Based Events", July 18, 2005
2. NEI White Paper, "Enhancements to Emergency Preparedness Programs for Hostile Action", November 18, 2005
3. PNPS 5.3.14, *"Security Incidents"*
4. PNPS 5.3.14.1, *"Airborne Threat"*

Site Area Emergency - HS4.1

Category: H - Hazards

Subcategory: 4 - Security

Initiating Condition: Hostile action within the Protected Area

EAL:

HS4.1 Site Area Emergency

A hostile action is occurring or has occurred within the **Protected Area** as reported by the Station Security Force

Mode Applicability:

All

NEI 99-01 Basis:

This condition represents an escalated threat to plant safety above that contained in the Alert in that a hostile force has progressed from the Owner Controlled Area to the Protected Area.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather, the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land, or water attack elements.

The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires offsite response organizations' (ORO) readiness and preparation for the implementation of protective measures.

This EAL addresses the potential for a very rapid progression of events due to a hostile action. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the Protected Area. Those events are adequately addressed by other EALs.

Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack.

PNPS Basis:

Security EALs could involve events due to nonhostile and hostile actions. It is important to understand the differences between these actions and therefore the definitions of hostile action and hostile force are provided with this basis.

Hostile Action: An act toward PNPS or its plant personnel that includes the use of violent force to destroy equipment, takes hostages, and/or intimidates plant personnel to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on PNPS. Nonterrorism-based EALs should be used to address such activities (e.g., violent acts between individuals in the Owner Controlled Area.)

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

This EAL is intended to address the potential for a very rapid progression of events due to an attack including:

- Air attack (aircraft impacting the Protected Area)
- Land-based attack (hostile force penetrating the Protected Area)
- Waterborne attack (hostile force on water penetrating the Protected Area)
- Bombs breaching the Protected Area

This EAL is intended to address the contingency for a very rapid progression of events due to a hostile attack and the possibility for additional attack. This EAL is not premised solely on the potential for a radiological release. Rather, the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional attack elements.

Site Area Emergency - HS4.1

This EAL addresses the immediacy of a threat to impact site vital areas within a relatively short time. The fact that the site is under serious attack with minimal time available for additional assistance to arrive requires offsite response organizations' readiness and preparation for the implementation of protective measures.

PNPS Basis Reference(s):

1. NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security Based Events", July 18, 2005
2. NEI White Paper, "Enhancements to Emergency Preparedness Programs for Hostile Action", November 18, 2005
3. PNPS 5.3.14, *"Security Incidents"*
4. PNPS 5.3.14.1, *"Airborne Threat"*

General Emergency - HG4.1

Category: H - Hazards

Subcategory: 4 - Security

Initiating Condition: Hostile action resulting in loss of physical control of the facility

EAL:

HG4.1 General Emergency

A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions

OR

A hostile action has caused failure of spent fuel cooling systems and imminent fuel damage is likely for recently irradiated fuel (fuel has decayed < 24 hours since reactor shutdown)

Mode Applicability:

All

NEI 99-01 Basis:

First Threshold

This EAL encompasses conditions under which a hostile action has resulted in a loss of 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.

If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the threshold is not met.

General Emergency - HG4.1

Second Threshold

This EAL addresses failure of spent fuel cooling systems as a result of hostile action if imminent fuel damage is likely, such as when a freshly off-loaded reactor core is in the spent fuel pool.

PNPS Basis:

The first threshold encompasses conditions under which a hostile force has taken control of plant equipment (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 (ability to cool the core), and decay heat removal (ability to maintain a heat sink). Temporary lineups to allow heat to be absorbed in the Torus are not considered a long term heat sink.

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. 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.

For the second threshold the term "recently irradiated fuel" is defined as fuel that has occupied part of a critical reactor core within the previous 24 hours (i.e., reactor fuel that has decayed less than 24 hours following reactor shutdown) (ref. 1).

PNPS Basis Reference(s):

1. Technical Specifications Section 3.7.C
2. PNPS 5.3.14, *"Security Incidents"*
3. PNPS 5.3.14.1, *"Airborne Threat"*

Alert - HA5.1

Category: H - Hazards

Subcategory: 5 - Control Room Evacuation

Initiating Condition: Control Room evacuation has been initiated

EAL:

HA5.1 Alert

PNPS 2.4.143, *"Shutdown from Outside Control Room,"* requires Control Room evacuation

Mode Applicability:

All

NEI 99-01 Basis:

With the Control Room evacuated, additional support, monitoring, and direction through the Technical Support Center and/or other emergency response facilities may be necessary.

Inability to establish plant control from outside the Control Room will escalate this event to a Site Area Emergency.

PNPS Basis:

PNPS 2.4.143, *"Shutdown from Outside Control Room,"* provides the instructions for scramming the unit and maintaining RCS inventory from outside the Control Room. The Shift Manager (SM) determines whether the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions.

PNPS Basis Reference(s):

1. PNPS 2.4.143, *"Shutdown from Outside Control Room"*

Category: H - Hazards

Subcategory: 5 - Control Room Evacuation

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

EAL:

HS5.1 Site Area Emergency

Control Room evacuation has been initiated

AND

Control of the plant **cannot** be established within 15 min.

Mode Applicability:

All

NEI 99-01 Basis:

The intent of this EAL is to capture those events where control of the plant cannot be re-established in a timely manner. In this case, expeditious transfer of control of safety systems has not occurred (although fission product barrier damage may not yet be indicated).

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 shut down the reactor and maintain it shutdown), reactor water level (ability to cool the core), and decay heat removal (ability to maintain a heat sink) I.

The determination of whether or not control is established at the remote shutdown panel is based on Emergency Director (ED) judgment. The Emergency Director is expected to make a reasonable, informed judgment within the site-specific time for transfer that the licensee has control of the plant from the remote shutdown panel.

Site Area Emergency - HS5.1

Escalation of this emergency classification level, if appropriate, would be by fission product barrier degradation or abnormal rad levels/radiological effluent EALs.

PNPS Basis:

PNPS 2.4.143, *"Shutdown from Outside Control Room,"* provides the instructions for scramming the unit and maintaining RPV inventory from outside the Control Room. The Shift Manager determines whether the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions.

PNPS Basis Reference(s):

1. PNPS 2.4.143, *"Shutdown from Outside Control Room"*

Unusual Event - HU6.1

Category: H - Hazards

Subcategory: 6 - Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a UE

EAL:

HU6.1 Unusual Event

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate **EITHER**:

A potential degradation of the level of safety of the plant

OR

A security threat to facility protection has been initiated

No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs

Mode Applicability:

All

NEI 99-01 Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the UE emergency classification level.

Unusual Event - HU6.1

PNPS Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the PNPS Emergency Plan. Initially, the Shift Manager or the available on-shift Senior Reactor Operator (SRO) assumes the duties and responsibilities as the Emergency Director. When augmentation of the on-shift complement occurs, the Senior Nuclear Executive or a designated alternate reports to the EOF and, once briefed, relieves the Shift Manager of all Emergency Director responsibilities. Once the on-call Emergency Director assumes the Emergency Director responsibilities, overall command and control of the emergency transfers from the Control Room to the EOF. The Emergency Plant Operations Supervisor (EPOS) may relieve the on-shift Emergency Director until such time as the on-call Emergency Director arrives; however, the EPOS must report and remain in the Control Room until relieved (ref. 1).

The "Judgment" classifications should not be used if an applicable EAL classification has already been determined as included in the EAL chart.

PNPS Basis Reference(s):

1. PNPS Emergency Plan Section B-3
2. Refer to PNPS EP-IP-100, *"Emergency Classification and Notification"*, for definition of "Unusual Event".

Alert - HA6.1

Category: H - Hazards

Subcategory: 6 - Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.

EAL:

HA6.1 Alert

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve **EITHER**:

An actual or potential substantial degradation of the level of safety of the plant

OR

A security event that involves probable life threatening risk to site personnel or damage to site equipment because of hostile action

Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels (1,000 mrem TEDE and 5,000 mrem thyroid CDE)

Mode Applicability:

All

NEI 99-01 Basis:

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

Alert - HA6.1

PNPS Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the PNPS Emergency Plan. Initially, the Shift Manager or the available on-shift Senior Reactor Operator (SRO) assumes the duties and responsibilities as the Emergency Director. When augmentation of the on-shift complement occurs, the Senior Nuclear Executive or a designated alternate reports to the EOF and, once briefed, relieves the Shift Manager of all Emergency Director responsibilities. Once the on-call Emergency Director assumes the Emergency Director responsibilities, overall command and control of the emergency transfers from the Control Room to the EOF. The Emergency Plant Operations Supervisor (EPOS) may relieve the on-shift Emergency Director until such time as the on-call Emergency Director arrives; however, the EPOS must report and remain in the Control Room until relieved (ref. 1).

The "Judgment" classifications should not be used if an applicable EAL classification has already been determined as included in the EAL chart.

PNPS Basis Reference(s):

1. PNPS Emergency Plan Section B-3
2. Refer to PNPS EP-IP-100, *"Emergency Classification and Notification"*, for definition of "Alert".

Site Area Emergency - HS6.1

Category: H - Hazards

Subcategory: 6 - Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency

EAL:

HS6.1 Site Area Emergency

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve **EITHER**:

An actual or likely major failure of plant functions needed for protection of the public

OR

Hostile action that results in intentional damage or malicious acts 1) toward site personnel or equipment that could lead to the likely failure of or 2) that prevent effective access to equipment needed for the protection of the public

Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1,000 mrem TEDE and 5,000 mrem thyroid CDE) beyond the site boundary

Mode Applicability:

All

NEI 99-01 Basis:

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

PNPS Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the PNPS Emergency Plan. Initially, the Shift Manager or the available on-shift Senior Reactor Operator (SRO) assumes the duties and responsibilities as the Emergency Director. When augmentation of the on-shift complement occurs, the Senior Nuclear Executive or a designated alternate reports to the EOF and, once briefed, relieves the Shift Manager of all Emergency Director responsibilities. Once the on-call Emergency Director assumes the Emergency Director responsibilities, overall command and control of the emergency transfers from the Control Room to the EOF. The Emergency Plant Operations Supervisor (EPOS) may relieve the on-shift Emergency Director until such time as the on-call Emergency Director arrives; however, the EPOS must report and remain in the Control Room until relieved (ref. 1).

The "Judgment" classifications should not be used if an applicable EAL classification has already been determined as included in the EAL chart.

PNPS Basis Reference(s):

1. PNPS Emergency Plan Section B-3
2. Refer to PNPS EP-IP-100, *"Emergency Classification and Notification"*, for definition of "Site Area Emergency".

General Emergency - HG6.1

Category: H - Hazards

Subcategory: 6 - Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency

EAL:

HG6.1 General Emergency

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve **EITHER**:

Actual or imminent substantial core degradation or melting with potential for loss of containment integrity

OR

Hostile action that results in an actual loss of physical control of the facility

Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1,000 mrem TEDE and 5,000 mrem thyroid CDE) beyond the site boundary

Mode Applicability:

All

NEI 99-01 Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for General Emergency.

General Emergency - HG6.1

PNPS Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the PNPS Emergency Plan. Initially, the Shift Manager or the available on-shift Senior Reactor Operator (SRO) assumes the duties and responsibilities as the Emergency Director. When augmentation of the on-shift complement occurs, the Senior Nuclear Executive or a designated alternate reports to the EOF and, once briefed, relieves the Shift Manager of all Emergency Director responsibilities. Once the on-call Emergency Director assumes the Emergency Director responsibilities, overall command and control of the emergency transfers from the Control Room to the EOF. The Emergency Plant Operations Supervisor (EPOS) may relieve the on-shift Emergency Director until such time as the on-call Emergency Director arrives; however, the EPOS must report and remain in the Control Room until relieved (ref. 1).

The "Judgment" classifications should not be used if an applicable EAL classification has already been determined as included in the EAL chart.

PNPS Basis Reference(s):

1. PNPS Emergency Plan Section B-3
2. Refer to PNPS EP-IP-100, *"Emergency Classification and Notification"*, for definition of "General Emergency".

Category S - System Malfunction

EAL Group: Hot Conditions (RCS temperature > 212°F); EALs in this category are applicable only in one or more hot operating modes.

Numerous system-related equipment failure events that warrant emergency classification have been identified in this category. They may pose actual or potential threats to plant safety.

The events of this category pertain to the following subcategories:

1. Loss of AC Power

Loss of emergency electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity.

2. ATWS/Criticality

Events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor scrams. In the plant licensing basis, postulated failures of the RPS to complete a reactor scram comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification however, ATWS is intended to mean any scram failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS, and Primary Containment integrity. Inadvertent criticalities pose potential personnel safety hazards as well being indicative of losses of reactivity control.

3. Inability to Reach Shutdown Conditions

One EAL falls into this subcategory. It is related to the failure of the plant to be brought to the required plant operating condition required by the Technical Specifications if a limiting condition for operation (LCO) is not met.

4. Instrumentation/Communications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Loss of annunciators or indicators is in this subcategory.

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

5. Fuel Clad Degradation

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant increase from these baseline levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specifications limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

6. RCS Leakage

The reactor vessel provides a volume for the coolant that covers the reactor core. The reactor vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail.

Excessive RCS leakage greater than Technical Specifications limits is utilized to indicate potential pipe cracks that may propagate to an extent threatening fuel clad, RCS, and Primary Containment integrity.

7. Loss of DC Power

Loss of emergency electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category involves total losses of vital plant 125V DC power sources.

Unusual Event - SU1.1

Category: S - System Malfunction

Subcategory: 1 - Loss of AC Power

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

EAL:

SU1.1 Unusual Event

Loss of **all** offsite AC power (Table S-3) to emergency buses A5 and A6 for ≥ 15 min. (Note 3)

Note 3: The Emergency Director should not wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table S-3 AC Power Sources
Offsite
- Startup Transformer (X4) - Shutdown Transformer - UAT - Backscuttle via Main Transformer (only if already established)
Onsite
- EDG A - EDG B - SBO DG

Unusual Event - SU1.1

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

Prolonged loss of offsite AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of AC power to emergency buses.

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

PNPS Basis:

The PNPS emergency 4160V AC electrical distribution system is illustrated in Figure S-1.

The inability to provide emergency AC power from any offsite power source poses a potential degradation of reactor plant safety. If onsite power capability becomes degraded, significant losses of vital equipment operability may occur. 4160V AC buses A5 and A6 are the emergency buses. These electrical buses provide power to vital equipment such as RHR, Core Spray, and CRD pumps as well as vital 480V AC transformers feeding buses B1 and B2. Offsite power supply transformers are those transformers which are capable of providing power to the emergency buses independent of the onsite generators. Due to the significant time required to establish the necessary switching lineups and removal of generator disconnect links, backscuttle is not considered an available offsite power supply transformer unless already established at the time of the loss of other offsite power supply transformers.

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. If neither emergency bus is energized by an offsite source within 15 minutes, an Unusual Event is declared under this EAL.

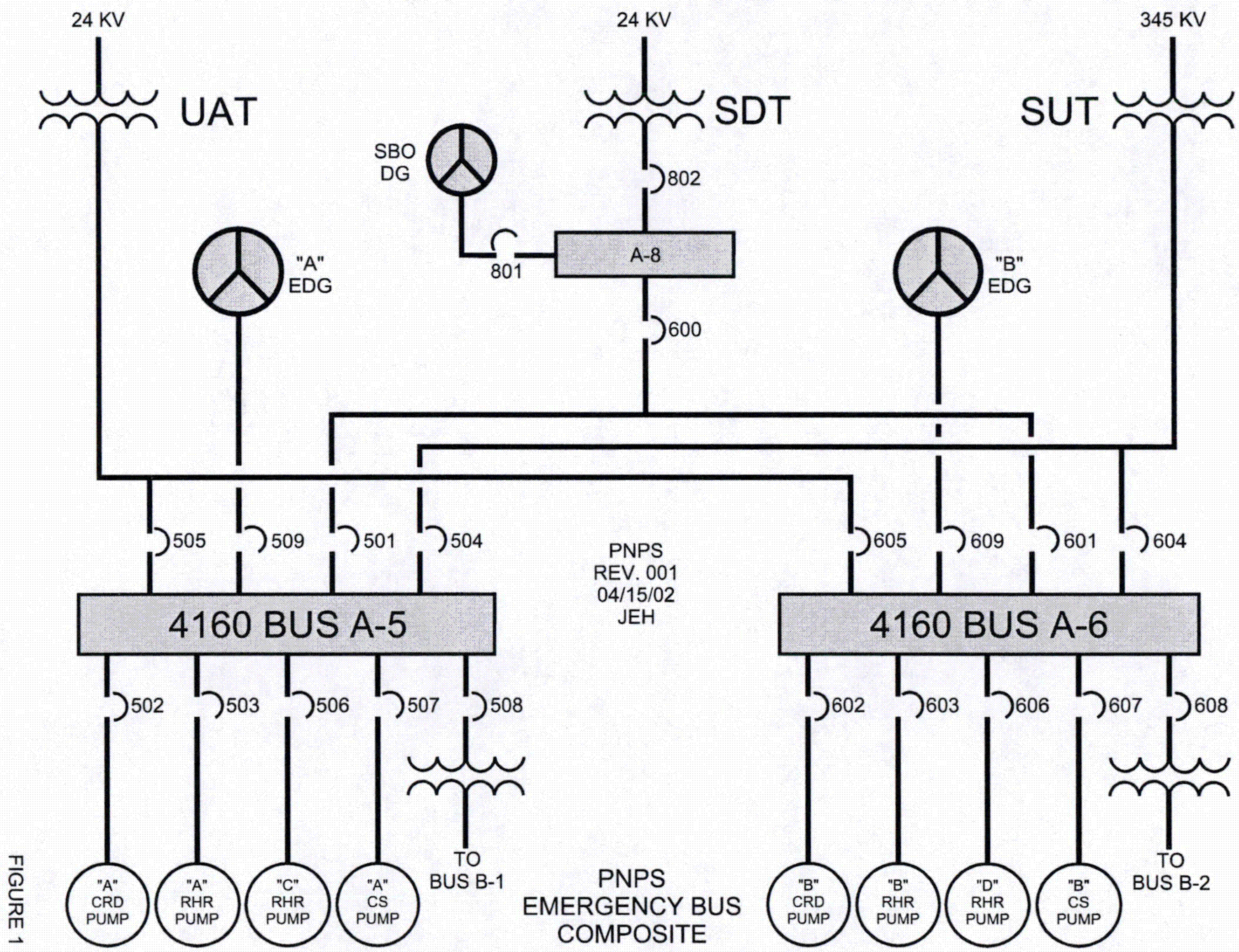
Unusual Event - SU1.1

PNPS Basis Reference(s):

1. Bechtel Drawing Electrical Single Line Diagram S-E-155
2. FSAR Section 8.1
3. FSAR Section 8.3
4. FSAR Section 8.5
5. PNPS 5.3.31, *"Station Blackout"*
6. PNPS 2.2.8, *"Standby AC Power System (Diesel Generators)"*
7. PNPS 2.2.146, *"Station Blackout Diesel Generator"*
8. PNPS 2.4.144, *"Degraded Voltage"*
9. PNPS 2.4.A.23, *"Loss/Degradation Of 23kV Line"*
10. PNPS 2.4.A.5, *"Loss of Electrical Bus A5"*
11. PNPS 2.4.A.6, *"Loss of Electrical Bus A6"*

Unusual Event - SU1.1

Figure S-1 Emergency Bus Composite Diagram



Alert - SA1.1

Category: S - System Malfunction

Subcategory: 1 - Loss of AC Power

Initiating Condition: AC power capability to emergency buses reduced to a single power source for 15 minutes or longer such that **any** additional single failure would result in loss of all AC power to emergency buses

EAL:

SA1.1 Alert

AC power capability to emergency buses A5 and A6 reduced to a single power source (Table S-3) for ≥ 15 min. such that **any** additional single failure would result in loss of **all** AC power to emergency buses (Note 3)

Note 3: The Emergency Director should **not** wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table S-3 AC Power Sources
Offsite
- Startup Transformer (X4) - Shutdown Transformer - UAT - Backscuttle via Main Transformer (only if already established)
Onsite
- EDG A - EDG B - SBO DG

Alert - SA1.1

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

The condition indicated by this EAL is the degradation of the offsite and onsite AC 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 all but one emergency generator to supply power to its emergency buses. Another related condition could be the loss of all offsite power and loss of onsite emergency generators with only one train of emergency buses being backfed from the unit main generator, or the loss of onsite emergency generators with only one train of emergency buses being backfed from offsite power. The subsequent loss of this single power source would escalate the event to a Site Area Emergency in accordance with EAL SS1.1.

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

PNPS Basis:

The PNPS emergency 4160V AC electrical distribution system is illustrated in Figure S-1.

The inability to provide emergency AC power from any offsite power source poses a potential degradation of reactor plant safety. If onsite power capability becomes degraded, significant losses of vital equipment operability may occur. 4160V AC buses A5 and A6 are the emergency buses. These electrical buses provide power to vital equipment such as RHR, Core Spray, and CRD pumps as well as vital 480V AC transformers feeding buses B1 and B2. Offsite power supply transformers are those transformers which are capable of providing power to the emergency buses independent of the onsite generators. Due to the significant time required to establish the necessary switching lineups and removal of generator disconnect links, backscuttle is not considered an available offsite power supply transformer unless already established at the time of the loss of other offsite power supply transformers.

PNPS Basis Reference(s):

1. Bechtel Drawing Electrical Single Line Diagram S-E-155
2. FSAR Section 8.1
3. FSAR Section 8.3
4. FSAR Section 8.5
5. PNPS 5.3.31, *"Station Blackout"*
6. PNPS 2.2.8, *"Standby AC Power System (Diesel Generators)"*
7. PNPS 2.2.146, *"Station Blackout Diesel Generator"*
8. PNPS 2.4.144, *"Degraded Voltage"*
9. PNPS 2.4.A.23, *"Loss/Degradation Of 23kV Line"*
10. PNPS 2.4.A.5, *"Loss of Electrical Bus A5"*
11. PNPS 2.4.A.6, *"Loss of Electrical Bus A6"*

Alert - SA1.1

Figure S-1 Emergency Bus Composite Diagram

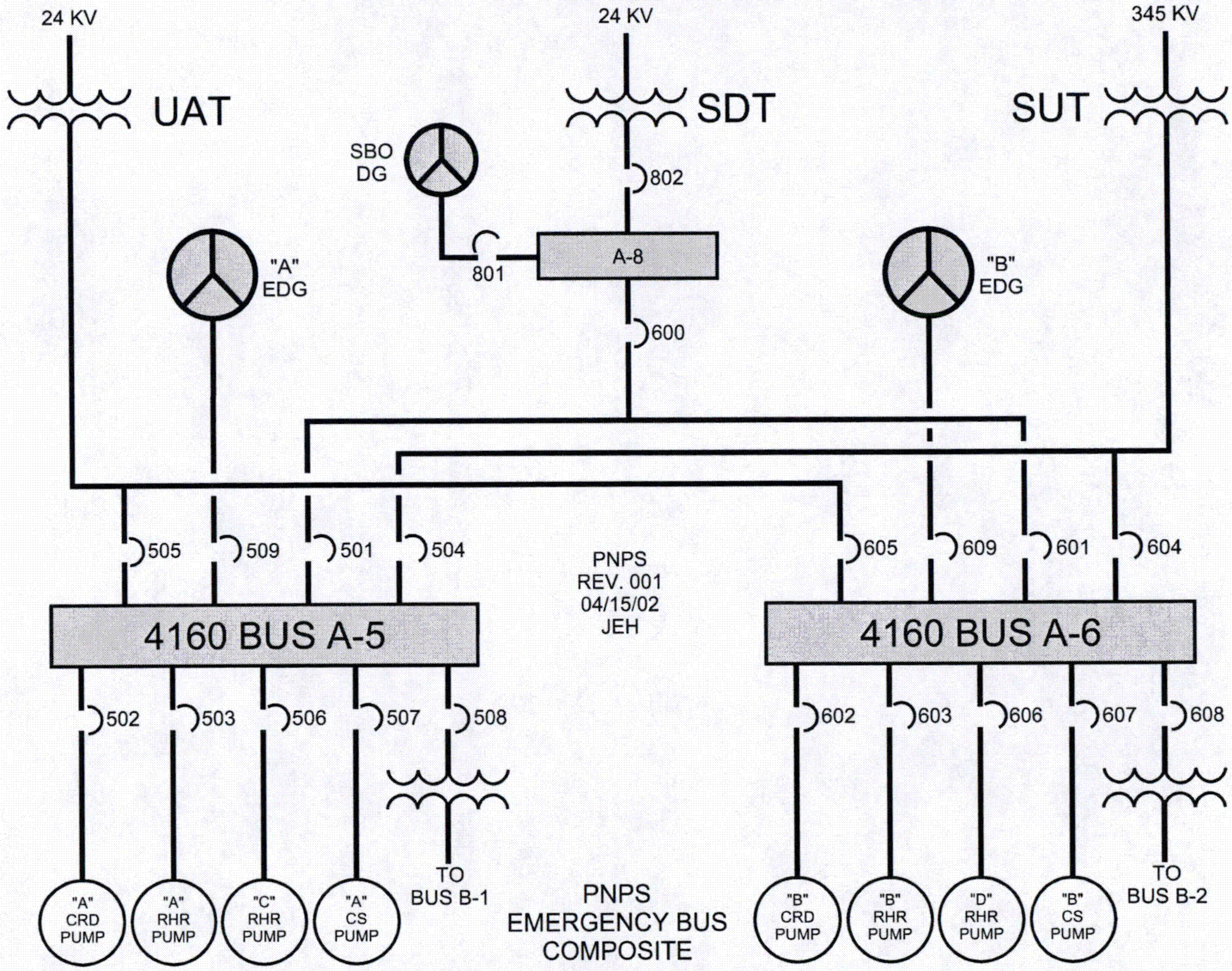


FIGURE 1

Site Area Emergency - SS1.1

Category: S - System Malfunction

Subcategory: 1 - Loss of AC Power

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

EAL:

SS1.1 Site Area Emergency

Loss of **all** offsite and **all** onsite AC power (Table S-3) to emergency buses A5 and A6 for ≥ 15 min. (Note 3)

Note 3: The Emergency Director should **not** wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table S-3 AC Power Sources
Offsite
- Startup Transformer (X4) - Shutdown Transformer - UAT - Backscuttle via Main Transformer (only if already established)
Onsite
- EDG A - EDG B - SBO DG

Site Area Emergency - SS1.1

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

Loss of all AC power to emergency buses compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, and the Ultimate Heat Sink. Prolonged loss of all AC power to emergency buses will lead to loss of fuel clad, RCS, and Primary Containment; thus, this event can escalate to a General Emergency.

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

Escalation to General Emergency is via fission product barrier degradation EALs or EAL SG1.1.

PNPS Basis:

The PNPS emergency 4160V AC electrical distribution system is illustrated in Figure S-1.

The inability to provide emergency AC power from any offsite power source poses a potential degradation of reactor plant safety. If onsite power capability becomes degraded, significant losses of vital equipment operability may occur. 4160V AC buses A5 and A6 are the emergency buses. These electrical buses provide power to vital equipment such as RHR, Core Spray, and CRD pumps as well as vital 480V AC transformers feeding buses B1 and B2. Offsite power supply transformers are those transformers which are capable of providing power to the emergency buses independent of the onsite generators. Due to the significant time required to establish the necessary switching lineups and removal of generator disconnect links, backscuttle is not considered an available offsite power supply transformer unless already established at the time of the loss of other offsite power supply transformers.

This EAL is the hot condition equivalent of the cold condition loss of all AC power EAL CA1.1. When in Cold Shutdown, Refuel, 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 buses relative to that existing when in hot conditions.

PNPS Basis Reference(s):

1. Bechtel Drawing Electrical Single Line Diagram S-E-155
2. FSAR Section 8.1
3. FSAR Section 8.3
4. FSAR Section 8.5
5. PNPS 5.3.31, *"Station Blackout"*
6. PNPS 2.2.8, *"Standby AC Power System (Diesel Generators)"*
7. PNPS 2.2.146, *"Station Blackout Diesel Generator"*
8. PNPS 2.4.144, *"Degraded Voltage"*
9. PNPS 2.4.A.23, *"Loss/Degradation Of 23kV Line"*
10. PNPS 2.4.A.5, *"Loss of Electrical Bus A5"*
11. PNPS 2.4.A.6, *"Loss of Electrical Bus A6"*

Site Area Emergency - SS1.1

Figure S-1 Emergency Bus Composite Diagram

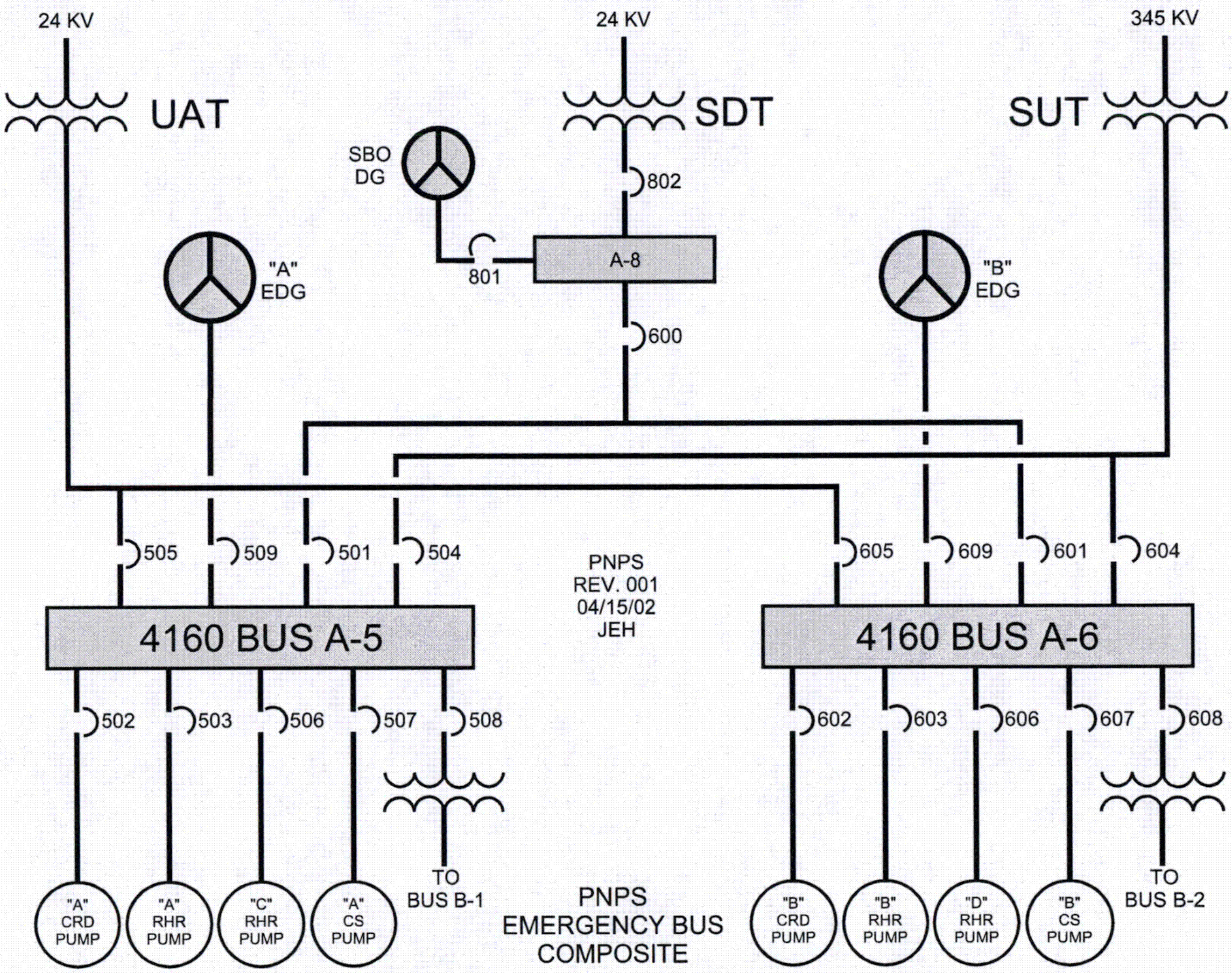


FIGURE 1

General Emergency - SG1.1

Category: S -System Malfunction

Subcategory: 1 - Loss of AC Power

Initiating Condition: Prolonged loss of **all** offsite power and prolonged loss of **all** onsite AC power to emergency buses

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power (Table S-3) to emergency buses A5 and A6

AND EITHER:

Restoration of at least one emergency bus in < 8 hours is **not** likely

OR

RPV level **cannot** be restored and maintained > -125 in. or **cannot** be determined

Table S-3 AC Power Sources
Offsite
- Startup Transformer (X4) - Shutdown Transformer - UAT - Backscuttle via Main Transformer (only if already established)
Onsite
- EDG A - EDG B - SBO DG

General Emergency - SG1.1

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

Loss of all AC power to emergency buses compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, and the Ultimate Heat Sink. Prolonged loss of all AC power to emergency buses will lead to loss of fuel clad, RCS, and containment, thus warranting declaration of a General Emergency.

This EAL is specified to assure that, in the unlikely event of a prolonged station blackout, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate based on a reasonable assessment of the event trajectory.

The likelihood of restoring at least one emergency bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

In addition, under these conditions, fission product barrier monitoring capability may be degraded.

PNPS Basis:

The PNPS emergency 4160V AC electrical distribution system is illustrated in Figure S-1.

The inability to provide emergency AC power from any offsite power source poses a potential degradation of reactor plant safety. Should onsite power capability become degraded, significant losses of vital equipment operability may occur. 4160V AC buses A5 and A6 are the emergency buses. These electrical buses provide power to vital equipment such as RHR, Core Spray, and CRD pumps as well as vital 480V AC transformers feeding buses B1 and B2. Offsite power supply transformers are those transformers which are capable of providing power to the emergency buses independent of the onsite generators. Due to the significant time required to establish the necessary switching lineups and removal of generator disconnect links, backscuttle is not considered an available offsite power supply transformer unless already established at the time of the loss of other offsite power supply transformers.

Indication of continuing core cooling degradation is manifested by an RPV level instrument reading of < -125 in. (RPV level is below TAF) (ref. 12). When RPV level is at or above TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme RPV level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops to TAF, the level is indicative of a challenge to core cooling and the fuel clad barrier.

When RPV level cannot be determined, EOPs require entry to EOP-16, *"RPV Flooding"*. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained (ref. 13). When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-16 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold the Minimum Steam Cooling Pressures (in scram-failure events). If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the fuel clad barrier exists.

PNPS Basis Reference(s):

1. Bechtel Drawing Electrical Single Line Diagram S-E-155
2. FSAR Section 8.1
3. FSAR Section 8.3
4. FSAR Section 8.5
5. PNPS 5.3.31, *"Station Blackout"*
6. PNPS 2.2.8, *"Standby AC Power System (Diesel Generators)"*
7. PNPS 2.2.146, *"Station Blackout Diesel Generator"*
8. PNPS 2.4.144, *"Degraded Voltage"*
9. PNPS 2.4.A.23, *"Loss/Degradation Of 23kV Line"*
10. PNPS 2.4.A.5, *"Loss of Electrical Bus A5"*
11. PNPS 2.4.A.6, *"Loss of Electrical Bus A6"*
12. EOP-1, *"RPV Control"*
13. EOP-16, *"RPV Flooding"*

General Emergency - SG1.1

Figure S-1 Emergency Bus Composite Diagram

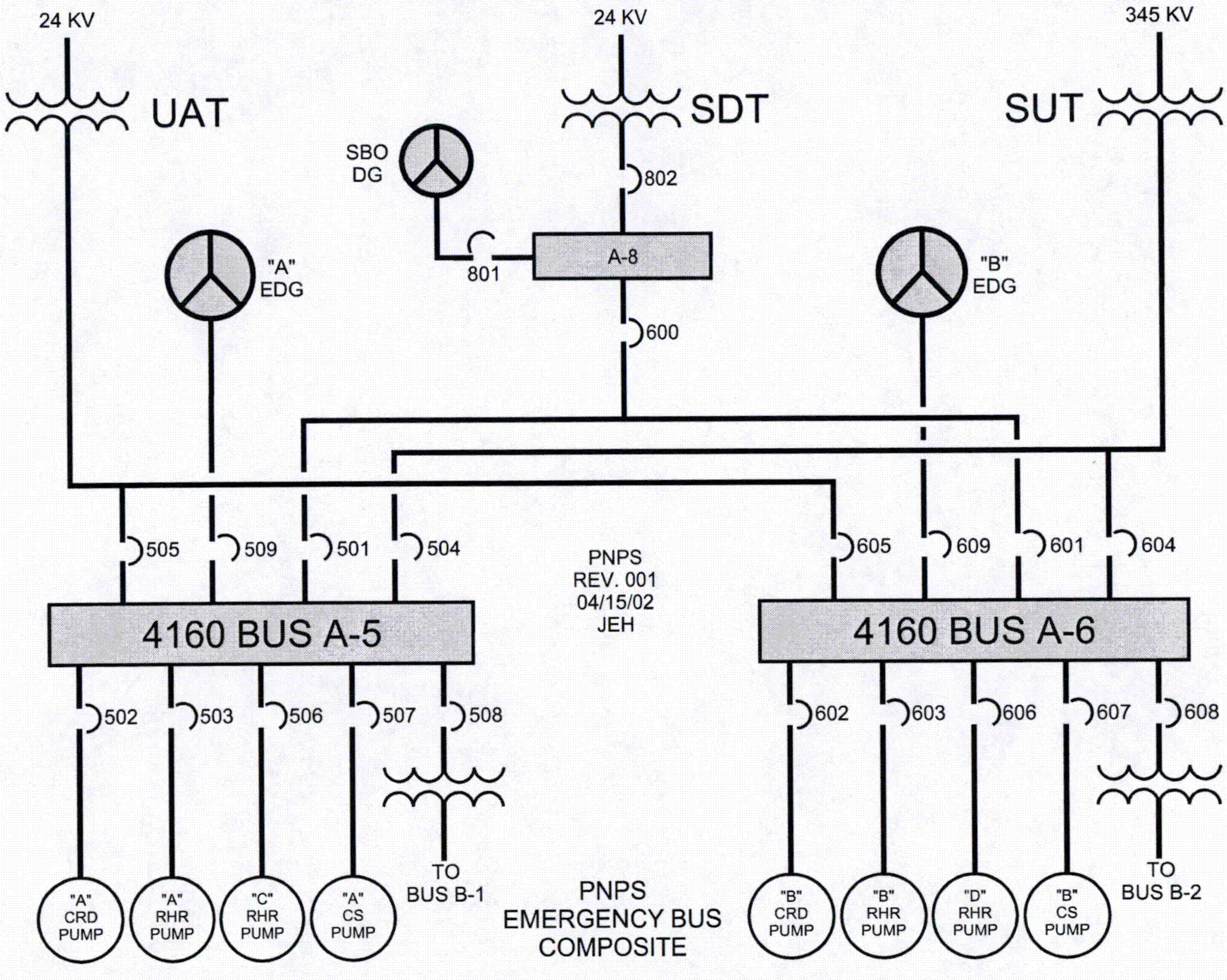


FIGURE 1

Unusual Event - SU2.1

Category: S - System Malfunction

Subcategory: 2 - ATWS/Criticality

Initiating Condition: Inadvertent criticality

EAL:

SU2.1	**Unusual Event**
Unplanned sustained positive period observed on nuclear instrumentation	

Mode Applicability:

2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

This EAL addresses inadvertent criticality events. This EAL indicates a potential degradation of the level of safety of the plant warranting a UE classification. This EAL excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated).

Escalation would be by the Fission Product Barrier Table, as appropriate to the operating mode at the time of the event.

PNPS Basis:

Period meters NI-750-4A, C, B, and D on Panel C905 identify this condition as well as annunciator **"SRM PERIOD"** (C905L-G9). Amber lights on Panel C905 illuminate when its SRM channel period is less than 20 seconds (seal in) (ref. 1, 2).

PNPS Basis Reference(s):

1. PNPS 2.2.64, *"Source Range Monitoring System"*
2. PNPS 3.M.2-5.1, *"Source Range Monitor Calibration Instruction"*

Alert - SA2.1

Category: S - System Malfunction

Subcategory: 2 - ATWS/Criticality

Initiating Condition: Automatic scram fails to shut down the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor.

EAL:

SA2.1 Alert

An automatic scram failed to shut down the reactor (reactor power < 3%)

AND

Manual actions taken at the reactor control console successfully shut down the reactor as indicated by reactor power < 3% (APRM downscale) (Note 6)

Note 6: Manual scram actions taken at the reactor control console are the following:

- Reactor Scram push buttons
- Reactor Mode switch in SHUTDOWN.
- ATWS-ARI push buttons.

Mode Applicability:

1 - Run, 2 - Startup

NEI 99-01 Basis:

Manual scram actions taken at the reactor control console are any set of actions by the reactor operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

Alert - SA2.1

This condition indicates failure of the automatic protection system to scram 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. Thus the plant safety has been compromised because design limits of the fuel may have been exceeded. An Alert is indicated because conditions may exist that lead to potential loss of fuel clad or RCS and because of the failure of the Reactor Protection System to automatically shut down the plant.

If manual actions taken at the reactor control console fail to shut down the reactor, the event would escalate to a Site Area Emergency.

PNPS Basis:

A reactor scram is automatically initiated by the Reactor Protection System (RPS) when certain continuously monitored parameters exceed predetermined setpoints. In Hot condition operating modes, a reactor scram may be the result of manual or automatic action in response to any of the following parameters (ref. 1):

- IRM High-High Flux
- IRM Inoperable
- APRM High-High Flux
- APRM High Flux (15%)
- APRM Inoperable
- APRM Downscale
- Rx Pressure High
- Drywell Pressure High
- Rx Water Level Low
- Scram Discharge Volume High
- Main Steam Line Isolation

Alert - SA2.1

- Turbine Control Valve Fast Closure
- Turbine Stop Valve Closure

Following a successful reactor scram, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative period. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-scram response from an automatic reactor scram signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful scram has therefore occurred when there is sufficient rod insertion to bring the reactor power below the APRM downscale setpoint of 3% (ref. 2).

This EAL indicates a failure of the automatic RPS scram function to rapidly insert a sufficient number of control rods to achieve reactor shutdown. The significance, therefore, is that a potential degradation of a safety system exists because a front line automatic protection system did not function in response to a plant transient. Thus, plant safety has been compromised.

Following any automatic RPS scram signal, PNPS 2.1.6 (ref. 3) prescribes insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved. Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Alert.

For the purpose of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console, Panel C905 (ref. 3, 4):

- Reactor Scram push buttons
- Reactor Mode switch in SHUTDOWN.
- ATWS-ARI push buttons.

Reactor shutdown achieved by use of the EOP satellite procedures referenced in EOP-2 do not constitute a successful manual scram (ref. 5).

This EAL is not applicable if a manual scram is initiated and no RPS setpoints are exceeded. Taking the mode switch to SHUTDOWN position is a manual scram action. When the mode switch is taken out of the RUN position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

In the event that the operator identifies a reactor scram is imminent and initiates a successful manual reactor scram before the automatic scram setpoint is reached, no declaration is required. The successful manual scram of the reactor before it reaches its automatic scram setpoint or reactor scram signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. If manual reactor scram actions fail to reduce reactor power below 3% (ref. 2), the event escalates to the Site Area Emergency under EAL SS2.1.

If, by procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded), it may be difficult to determine whether the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, consideration should be given to evaluating the fuel for potential damage and the reporting requirements of 50.72 should be considered for the transient event.

PNPS Basis Reference(s):

1. PNPS 2.2.79, *"Reactor Protection System"*
2. EOP-1, *"RPV Control"*
3. PNPS 2.1.6, *"Reactor Scram"*
4. PNPS 2.2.126, *"Anticipated Transient Without Scram (ATWS)"*
5. EOP-2, "*RPV Control, Failure-to-Scram*"

Site Area Emergency - SS2.1

Category: S - System Malfunction

Subcategory: 2 - ATWS/Criticality

Initiating Condition: Automatic scram fails to shut down the reactor and manual actions taken from the reactor control console are **not** successful in shutting down the reactor

EAL:

SS2.1 Site Area Emergency

An automatic scram failed to shut down the reactor (reactor power < 3%)

AND

Manual actions taken at the reactor control console do not shut down the reactor as indicated by reactor power ≥ 3% (Note 6)

Note 6: Manual scram actions taken at the reactor control console are the following:

- Reactor Scram push buttons
- Reactor Mode switch in SHUTDOWN.
- ATWS-ARI push buttons.

Mode Applicability:

1 - Run, 2 - Startup

Site Area Emergency - SS2.1

NEI 99-01 Basis:

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful. A Site Area Emergency is warranted because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS.

Manual scram actions taken at the reactor control console are any set of actions by the reactor operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

Manual scram actions are not considered successful if action away from the reactor control console is required to scram (trip) the reactor. This EAL is still applicable even if actions taken away from the reactor control console are successful in shutting down the reactor because the design limits of the fuel may have been exceeded or because of the gross failure of the Reactor Protection System to shut down the plant.

Escalation of this event to a General Emergency would be due to a prolonged condition leading to an extreme challenge to either core cooling or heat removal.

PNPS Basis:

This EAL addresses any automatic reactor scram signal followed by a manual scram that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed. For the purpose of emergency classification at the Site Area Emergency level, successful manual scram actions are those which can be quickly performed from the reactor control console, Panel C905 (ref. 1, 2):

- Reactor Scram push buttons
- Reactor Mode switch in SHUTDOWN.
- ATWS-ARI push buttons.

Reactor shutdown achieved by use of the EOP satellite procedures referenced in EOP-2 do not constitute a successful manual scram (ref. 3).

Site Area Emergency - SS2.1

The APRM downscale trip setpoint is a minimum reading on the power range scale that indicates power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, torus temperature trend) can be used to determine whether reactor power is greater than 3% power (ref. 4).

Escalation of this event to a General Emergency would be under EAL SG2.1 or Emergency Director judgment.

PNPS Basis Reference(s):

1. PNPS 2.1.6, *"Reactor Scram"*
2. PNPS 2.2.126, *"Anticipated Transient Without Scram (ATWS)"*
3. EOP-2, *"RPV Control, Failure-to-Scram"*
4. EOP-1, *"RPV Control"*

General Emergency - SG2.1

Category: S - System Malfunction

Subcategory: 2 - ATWS/Criticality

Initiating Condition: Failure of the RPS 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

EAL:

SG2.1 General Emergency

An automatic scram failed to shut down the reactor (reactor power < 3%)

AND

All manual actions do not shut down the reactor as indicated by reactor power ≥ 3%

AND EITHER: of the following exists or has occurred due to continued power generation:

RPV level **cannot** be restored and maintained > -150 in. or **cannot** be determined

OR

Torus water temperature and RPV pressure **cannot** be maintained below Heat Capacity Temperature Limit (EOP-11 Figure 2)

Mode Applicability:

1 - Run, 2 - Startup

NEI 99-01 Basis:

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful.

In the event either of these challenges exists 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 table declaration to permit maximum offsite intervention time.

PNPS Basis:

This EAL addresses the following:

- Any automatic reactor scram signal followed by failure of the automatic scram and all subsequent manual scrams to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed (EAL SS2.1); and

- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

In addition to manual scram methods at the reactor control console (Panel C905), a reactor shutdown achieved by use of the EOP support procedures referenced in EOP-2 is also credited as a successful manual scram provided that reactor power can be reduced below the APRM downscale trip setpoint before indications of an extreme challenge to either core cooling or heat removal exist (ref. 1).

The APRM downscale trip setpoint is a minimum reading on the power range scale that indicates power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, torus temperature trend) can be used to determine whether reactor power is greater than 2% power (ref. 2).

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the fuel clad and RCS barriers.

Indication that core cooling is extremely challenged is manifested by inability to restore and maintain RPV water level above -150 in. (or cannot be determined). -150 in. is the Minimum Steam Cooling RPV Water Level (MSCRWL). The MSCRWL is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500°F. This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core spray cooling could be a precursor of a core melt sequence. When RPV level cannot be determined under ATWS conditions, EOPs require entry to EOP-26, *"RPV Flooding, Failure-to-Scram"* (ref. 3). RPV water level indication provides the primary means of knowing whether adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-26 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold the Minimum Steam Cooling Pressures.

The Heat Capacity Temperature Limit (HCTL) is the highest torus temperature from which emergency RPV depressurization will not raise torus pressure above the Primary Containment Pressure Limit (PCPL) while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent. The HCTL is a function of RPV pressure and torus level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant. This threshold is met when EOP-3, *"Primary Containment Control"*, Step TT-10 is reached (ref. 4). This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature.

General Emergency - SG2.1

It is recognized that actions will be taken in the plant Emergency Operating Procedures (EOPs) in response to the failure to scram. The conditions addressed in the EAL thresholds ("RPV level **cannot** be restored and maintained > -150 in..." and "torus water temperature and RPV pressure **cannot** be maintained below the Heat Capacity Temperature Limit...") correspond to decisions made in the EOPs. If at any time it is determined that RPV level cannot be restored and maintained > -150 in. or torus water temperature and RPV pressure cannot be maintained below the Heat Capacity Temperature Limit, a General Emergency must be declared.

PNPS Basis Reference(s):

1. EOP-2, *"RPV Control, Failure-to-Scram"*
2. EOP-1, *"RPV Control"*
3. EOP-26, *"RPV Flooding, Failure-to-Scram"*
4. EOP-3, *"Primary Containment Control"*
5. PNPS 2.1.6, *"Reactor Scram"*
6. PNPS 2.2.126, *"Anticipated Transient Without Scram (ATWS)"*

Unusual Event - SU3.1

Category: S - System Malfunction

Subcategory: 3 - Inability to Reach Shutdown Conditions

Initiating Condition: Inability to reach required shutdown within Technical Specifications limits

EAL:

SU3.1 Unusual Event

Plant is **not** brought to required operating mode within Technical Specifications LCO action statement time

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

Limiting Conditions for Operation (LCOs) require the plant to be brought to a required operating mode when the Technical Specifications required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the site Technical Specifications requires a 4-hour report under 10CFR50.72(b) Nonemergency 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 PNPS Technical Specifications and is not related to how long a condition may have existed.

PNPS Basis:

None

Unusual Event - SU3.1

PNPS Basis Reference(s):

1. PNPS Technical Specifications

Unusual Event - SU4.1

Category: S - System Malfunction

Subcategory: 4 - Instrumentation/Communications

Initiating Condition: Unplanned loss of safety system annunciation or indication in the Control Room for 15 minutes or longer

EAL:

SU4.1 Unusual Event

Unplanned loss of > approximately 75% of annunciators or indicators associated with safety systems on MCR Panels C903, C904, C905, C1, C3, C170, and C171 for ≥ 15 min. (Note 3)

Note 3: The Emergency Director should **not** wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

This EAL is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer-based indication equipment is considered (e.g., Process Computer, EPIC, SPDS, etc.).

Loss of annunciators or indicators excludes scheduled maintenance and testing activities.

Quantification 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 provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specifications. The initiation of a Technical Specifications imposed plant shutdown related to the instrument loss will be reported via 10CFR50.72. If the shutdown is not in compliance with the Technical Specifications action, the UE is based on EAL SU3.1.

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

This UE will be escalated to an Alert based on a concurrent loss of compensatory indications or if a significant transient is in progress during the loss of annunciation or indication.

PNPS Basis:

The availability of computer-based monitoring capability (i.e., Process Computer, EPIC, SPDS) is a factor at the Alert classification level but not a factor at the Unusual Event emergency classification level. Safety system annunciation and indication considered in this EAL are found on MCR Panels C903, C904, C905, C1, C3, C170, and C171. The other annunciators and indicators are important to plant operation but are not important to safety (ref. 1, 2).

PNPS Basis Reference(s):

1. PNPS 2.3.1, *"General Action for Alarm Response and Annunciator Control"*
2. PNPS 2.4.155, *"Loss of Annunciator System"*

Unusual Event - SU4.2

Category: S - System Malfunction

Subcategory: 4 - Instrumentation/Communications

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

EAL:

SU4.2 Unusual Event

Loss of **all** Table S-2 onsite (internal) communication methods affecting the ability to perform routine operations

OR

Loss of **all** Table S-2 offsite (external) communication methods affecting the ability to perform offsite notifications

Table S-2 Communications Systems

System	Onsite (internal)	Offsite (external)
Plant Telephone System (CENTREX)	X	X
Wireless Telephone System	X	X
Pilgrim Station Radio System	X	
Plant Gaitronics System	X	
Alternate Shutdown Communication	X	
NRC-ENS Telephone, Direct Line		X
Satellite phones		X

Unusual Event - SU4.2

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

The purpose of this EAL 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 issues with offsite authorities.

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

PNPS Basis:

Onsite/offsite communications include one or more of the systems listed in Table S-2 (ref. 1, 2, 3, 4, 5). A description of the capabilities of each system is given in Section 4.0 of PNPS 2.2.17, *"Communications Systems"* (ref. 2).

This EAL is the hot condition equivalent of the cold condition EAL CU4.1.

PNPS Basis Reference(s):

1. FSAR Section 10.15
2. PNPS 2.2.17, *"Communications Systems"*
3. PNPS 2.4.57, *"Loss of Public-Address System"*
4. PNPS 8.A.13, *"Plant Emergency Alarms and Radio Test"*
5. EP-AD-413, *"Emergency Communications Test"*

Alert - SA4.1

Category: S - System Malfunction

Subcategory: 4 - Instrumentation/Communications

Initiating Condition: Unplanned loss of safety system annunciation or indication in the Control Room with **EITHER** 1) a significant transient in progress or 2) compensatory indicators unavailable

EAL:

SA4.1 **Alert**

Unplanned loss of > approximately 75% of annunciators or indicators associated with safety systems on MCR Panels C903, C904, C905, C1, C3, C170, and C171 for ≥ 15 min. (Note 3)

AND EITHER:

Any significant transient is in progress, Table S-1

OR

Compensatory indications are unavailable

Note 3: The Emergency Director should **not** wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table S-1 Significant Transients
Reactor scram
Runback > 25% thermal power
Electrical load rejection > 25% full electrical load
ECCS injection
Thermal power oscillations > 10%

Alert - SA4.1

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

This EAL is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a significant transient.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification is arbitrary; however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Supervisor be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specifications. The initiation of a Technical Specifications imposed plant shutdown related to the instrument loss will be reported via 10CFR50.72. If the shutdown is not in compliance with the Technical Specifications action, the UE is based on EAL SU3.1, "Inability to Reach Required Shutdown Within Technical Specifications Limits."

"Compensatory indications" in this context include computer-based information such as SPDS, Process Computer, EPIC, etc. If both a major portion of the annunciation system and all computer monitoring are unavailable, the Alert is required.

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

Alert - SA4.1

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress due to a concurrent loss of compensatory indications with a significant transient in progress during the loss of annunciation or indication.

PNPS Basis:

The Process Computer, EPIC, or SPDS serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators. Safety system annunciation and indication considered in this EAL are found on MCR Panels C903, C904, C905, C1, C3, C170, and C171. The other annunciators and indicators are important to plant operation but are not important to safety (ref. 1, 2).

Significant transients are listed in Table S-1 and include response to automatic or manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, electrical load rejections of greater than 25% full electrical load, ECCS injections, or thermal power oscillations of 10% or greater.

PNPS Basis Reference(s):

1. PNPS 2.3.1, *"General Action for Alarm Response and Annunciator Control"*
2. PNPS 2.4.155, *"Loss of Annunciator System"*

Site Area Emergency - SS4.1

Category: S - System Malfunction

Subcategory: 4 - Instrumentation/Communications

Initiating Condition: Inability to monitor a significant transient in progress

EAL:

SS4.1 Site Area Emergency

Loss of > approximately 75% of the annunciators or indicators associated with safety systems on MCR Panels C903, C904, C905, C1, C3, C170, and C171 for ≥ 15 min. (Note 3)

AND

Any significant transient is in progress, Table S-1

AND

Compensatory nonalarming indications are unavailable

Note 3: The Emergency Director should **not** wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table S-1 Significant Transients
Reactor scram
Runback > 25% thermal power
Electrical load rejection > 25% full electrical load
ECCS injection
Thermal power oscillations > 10%

Site Area Emergency - SS4.1

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

This EAL is intended to recognize the threat to plant safety associated with the complete loss of capability of the Control Room staff to monitor plant response to a significant transient.

"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 is arbitrary; however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Supervisor be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specifications. The initiation of a Technical Specifications imposed plant shutdown related to the instrument loss will be reported via 10CFR50.72. If the shutdown is not in compliance with the Technical Specifications action, the UE is based on EAL SU3.1, "Inability to Reach Required Shutdown Within Technical Specifications Limits."

A Site Area Emergency is considered to exist if the Control Room staff cannot monitor safety functions needed for protection of the public while a significant transient is in progress.

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.

"Compensatory indications" in this context include computer-based information such as SPDS, Process Computer, EPIC, etc. This should include all computer systems available for this use depending on specific plant design and subsequent retrofits.

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

PNPS Basis:

The availability of computer-based monitoring capability (i.e., Process Computer, EPIC, SPDS) is a factor at the Site Area Emergency classification level because they are compensatory nonalarming indications. These serve as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators. Safety system annunciation and indication considered in this EAL are found on MCR Panels C903, C904, C905, C1, C3, C170, and C171. The other annunciators and indicators are important to plant operation but are not important to safety (ref. 1, 2).

The ability to monitor EOP parameters is ultimately necessary for protection of the health and safety of the public. These parameters include those used to determine such functions as the ability to shut down the reactor, to maintain the core cooled and in a coolable geometry, to remove heat from the core, to maintain the reactor coolant system intact, and to maintain containment integrity.

Significant transients are listed in Table S-1 and include response to automatic or manually initiated functions such as trips, runbacks involving greater than 25% thermal power change, electrical load rejections of greater than 25% full electrical load, ECCS injections, or thermal power oscillations of 10% or greater.

Due to the limited number of safety systems in operation during cold shutdown, refuel, and defueled modes, this EAL is not applicable during these modes of operation.

PNPS Basis Reference(s):

1. PNPS 2.3.1, *"General Action For Alarm Response And Annunciator Control"*
2. PNPS 2.4.155, *"Loss of Annunciator System"*
3. PNPS Plant-Specific and Severe Accident Management Guidelines

Unusual Event - SU5.1

Category: S - System Malfunction

Subcategory: 5 - Fuel Clad Degradation

Initiating Condition: Fuel clad degradation

EAL:

SU5.1 Unusual Event

Air Ejector Offgas Radiation Monitors RM-1705-3A and B (Panel C910) reading > Hi-Hi alarm for > 13 min.

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

This EAL is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

Escalation of this EAL to the Alert level is via the fission product barriers.

This threshold addresses PNPS radiation monitor readings that provide indication of a degradation of fuel clad integrity.

PNPS Basis:

The Air Ejector Offgas Radiation Monitors 1705-3A and B continuously monitor air ejector offgas radiation levels, initiate an alarm if gamma radiation levels exceed short-term maximum release rates, and initiate auto-closure of offgas isolation valves if these limits are exceeded for longer than 13 minutes. The monitors serve to detect fuel damage since noble gases leaking from fuel elements pass through the monitoring instruments prior to release.

The high-high trip and alarm are set to preclude exceeding the Technical Specifications offgas release limit of 500,000 µCi/sec noble gas (referenced to a 30-minute hold-up) (ref. 1).

Unusual Event - SU5.1

The timer associated with the protective function of the offgas radiation monitors allows the operator time to clear the initiating signal through power reduction. The inability to clear the initiating signal within the 13 minutes indicates a significant fuel cladding integrity problem which, in turn, is indicative of abnormal core conditions and a potential for increased radiological hazards in plant. An air ejector offgas radiation monitor hi-hi alarm which does not clear within 13 minutes therefore warrants declaration of an Unusual Event.

In the Hot modes, a steam source is available from which noncondensable gases can be separated for processing by the offgas system. The cold shutdown, refuel, and defueled modes do not afford a transfer mechanism from which the Air Ejector Offgas radiation monitors can draw a valid sample. The radiation monitors lose a valid sample source when the air ejectors are not in service (ref. 1).

PNPS Basis Reference(s):

1. PNPS ODCM Section 8.3 Steam Jet Air Ejector Monitor

Unusual Event - SU5.2

Category: S - System Malfunction

Subcategory: 5 - Fuel Clad Degradation

Initiating Condition: Fuel clad degradation

EAL:

SU5.2 Unusual Event

Reactor coolant system sample activity > 20 μCi/mL total iodine

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

This EAL is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

Escalation of this EAL to the Alert level is via the fission product barriers.

This threshold addresses coolant samples exceeding coolant Technical Specifications for transient iodine spiking limits.

PNPS Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. This EAL addresses reactor coolant samples exceeding Technical Specifications LCOs 3.6.B which are applicable in Hot operating modes (ref. 1). A radioactivity concentration of 20 μCi/mL total iodine can be reached if there is fuel cladding failure or if there is a failure or a prolonged shutdown of the cleanup demineralizer.

PNPS Basis Reference(s):

1. PNPS Technical Specifications 3.6.B

Unusual Event - SU6.1

Category: S - System Malfunction

Subcategory: 6 - RCS Leakage

Initiating Condition: RCS leakage

EAL:

SU6.1 Unusual Event

Unidentified or pressure boundary leakage > 10 gpm

OR

Identified leakage > 25 gpm (Note 7)

Note 7: See Table F-1, Fission Product Barrier Matrix, for possible escalation above the Unusual Event due to RCS Leakage

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

This EAL 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 or 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).

Relief valve normal operation should be excluded from this EAL. However, a relief valve that operates and fails to close according to design should be considered applicable to this EAL if the relief valve cannot be isolated.

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 EAL to the Alert level is via fission product barrier degradation EALs.

Unusual Event - SU6.1

PNPS Basis:

The following alarms may be indicative of a reactor coolant leak inside containment (ref. 1):

- **"C19A/B TROUBLE"** (C904LC-B3)
- **"DRYWELL EQPT DRAIN SUMP DISCH HIGH TOTAL FLOW"** (C20C-A3)
- **"DRYWELL FLOOR DRAIN SUMP DISCH HIGH TOTAL FLOW"** (C20C-B3)
- Any Drywell cooler leaking alarm on Panel C7L

Identified leakage is:

- Reactor coolant leakage into Drywell collection systems such as pump seal or valve packing leaks that is captured and conducted to a sump or collecting tank; or
- Reactor coolant leakage into the Drywell atmosphere from sources which are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be pressure boundary leakage.

Pressure boundary leakage is leakage through a nonisolable fault in a reactor coolant system component body, pipe wall, or vessel wall.

Unidentified leakage is all reactor coolant leakage which is not identified leakage (ref. 2).

PNPS Basis Reference(s):

1. PNPS 2.2.77, *"Drywell Leak Detection Systems"*
2. PNPS Technical Specifications 3.6.C

Site Area Emergency - SS7.1

Category: S - System Malfunction

Subcategory: 7 - Loss of DC Power

Initiating Condition: Loss of **all** essential DC power for 15 minutes or longer

EAL:

SS7.1	**Site Area Emergency**
< 105V DC bus voltage indications on **all** essential 125V DC buses (Panels D16 and D17) for ≥ 15 min. (Note 3)	

Note 3: The Emergency Director should **not** wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 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.

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

Escalation to a General Emergency would occur by EALs in Category A or Category F.

Site Area Emergency - SS7.1

PNPS Basis:

The essential 125V DC power distribution is illustrated in Figure S-2.

The loss of required essential 125V DC power poses a significant threat to decay heat removal capability and reactor plant safety. Operability of safety-related equipment and safety system protective functions are severely degraded. The 125V DC system provides power to CSCS initiation logics, controllers, and indications. The 125V DC system also provides control power and tripping power for high voltage AC protective devices. If not restored within a short period of time, significant system and equipment failures may be imminent depending upon plant conditions at the time of the loss.

Annunciators **"A 125V DC UNDERVOLTAGE"** (C3RC-A7) and **"B 125V DC UNDERVOLTAGE"** (C3RC-B7) alarm at 124V DC (decreasing) and signal loss of Panel D16 and D17, respectively.

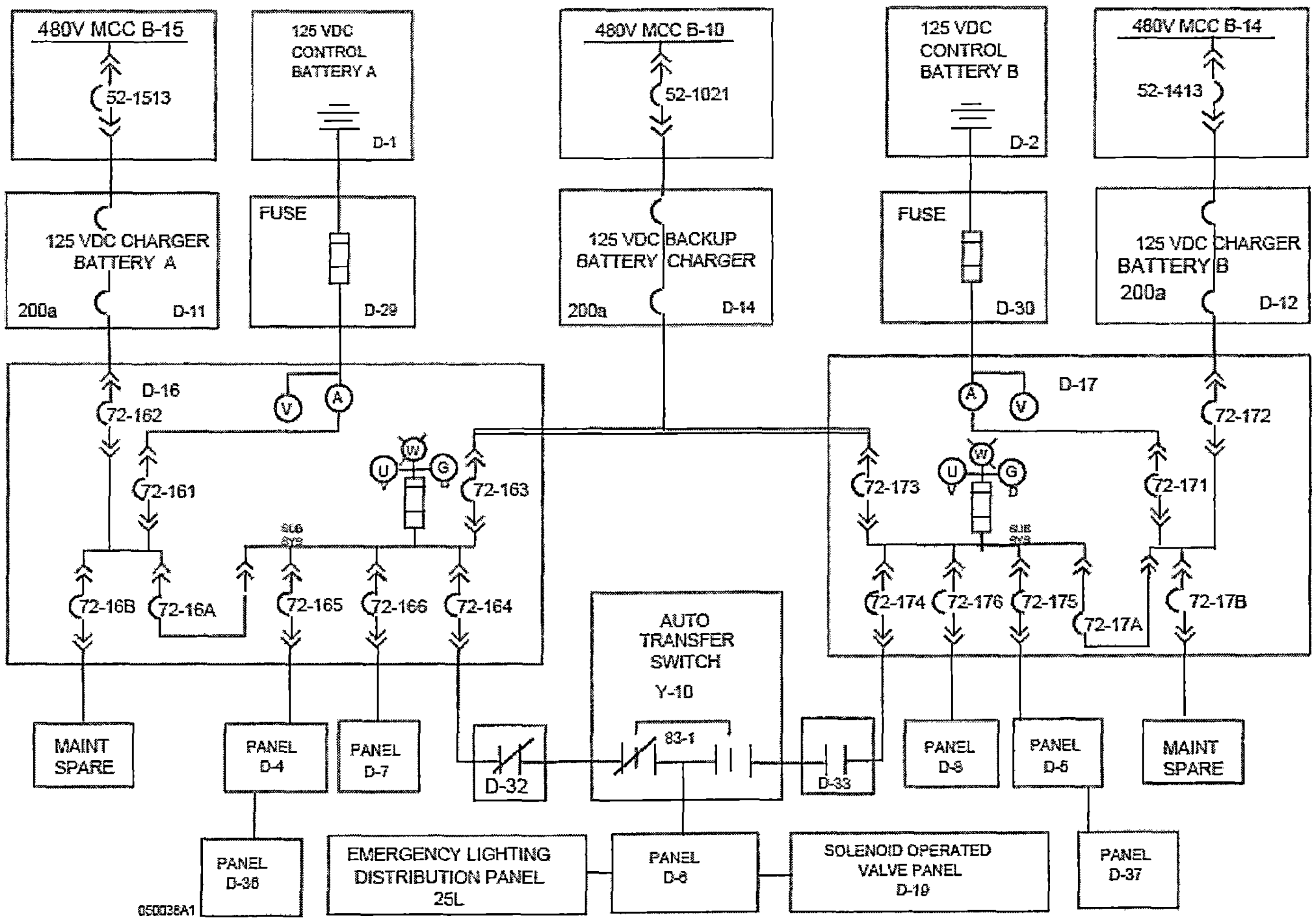
This EAL is the hot condition equivalent of the cold condition loss of DC power EAL CU6.1.

PNPS Basis Reference(s):

1. FSAR Figure 8.6-1
2. PNPS 2.2.14, *"125V DC Battery Systems"*
3. PNPS 5.3.11, *"Loss of Essential DC Bus D16 or D4 and D36"*
4. PNPS 5.3.12, *"Loss of Essential DC Bus D17 or D5 and D37"*
5. PNPS ARP C3RC-A7
6. PNPS ARP C3RC-B7

Site Area Emergency - SS7.1

Figure S-2 Essential 125V DC Power Distribution



Category F - Fission Product Barrier Degradation

EAL Group: Hot Conditions (RCS temperature > 212°F); EALs in this category are applicable only in one or more hot operating modes including Startup/Hot Standby.

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

A. Fuel Clad (FC): The fuel clad barrier consists of the zircalloy fuel bundle tubes that contain the fuel pellets.

B. Reactor Coolant System (RCS): 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.

C. Primary Containment (PC): The Primary Containment barrier includes the drywell, the wetwell (torus), their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 9.6). "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. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Unusual Event:

Any loss or any potential loss of Primary Containment

Alert:

Any loss or any potential loss of either fuel clad or RCS

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

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

- The fuel clad barrier and the RCS barrier are weighted more heavily than the Primary Containment barrier. UE EALs associated with RCS and fuel clad barriers are addressed under system malfunction EALs.
- At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if fuel clad and RCS barrier "loss" EALs existed, that, in addition to offsite dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both fuel clad and RCS barrier "potential loss" EALs existed, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.
- The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.
- The Primary Containment barrier should not be declared lost or potentially lost based on exceeding Technical Specifications action statement criteria unless there is an event in progress requiring mitigation by the Primary Containment barrier. When no event is in progress (Loss or Potential Loss of either fuel clad and/or RCS), the Primary Containment barrier status is addressed by Technical Specifications

Determine which combinations of the three barriers are lost or have a potential loss and use FU1.1, FA1.1, FS1.1, and FG1.1 to classify the event. Also, multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is imminent. In this imminent loss situation use judgment and classify as if the thresholds are exceeded.

Unusual Event - FU1.1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **Any** loss or **any** potential loss of Primary Containment

EAL:

FU1.1	**Unusual Event**
Any loss or **any** potential loss of Primary Containment (Table F-1, Attachment 9.6)	

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

None

PNPS Basis:

Fuel clad, RCS, and Primary Containment comprise the fission product barriers. Table F-1 (Attachment 9.6) lists the fission product barrier thresholds, bases, and references.

Fuel clad and RCS barriers are weighted more heavily than the Primary Containment barrier. Unlike the fuel clad and RCS barriers, the loss of either of which results in an Alert (EAL FA1.1), loss of the Primary Containment barrier in and of itself does not result in the relocation of radioactive materials or the potential for degradation of core cooling capability. However, loss or potential loss of the Primary Containment barrier in combination with the loss or potential loss of either the fuel clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1.

PNPS Basis Reference(s):

None

Alert - FA1.1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **Any** loss or **any** potential loss of either fuel clad or RCS

EAL:

FA1.1 Alert

Any loss or **any** potential loss of either fuel clad or RCS (Table F-1, Attachment 9.6)

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

None

PNPS Basis:

Fuel clad, RCS, and Primary Containment comprise the fission product barriers. Table F-1 (Attachment 9.6) lists the fission product barrier thresholds, bases, and references.

At the Alert classification level, fuel clad and RCS barriers are weighted more heavily than the Primary Containment barrier. Unlike the Primary Containment barrier, loss or potential loss of either the fuel clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Primary Containment barrier in combination with loss or potential loss of either fuel clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1.

PNPS Basis Reference(s):

None

Site Area Emergency - FS1.1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss or potential loss of **any** two barriers

EAL:

FS1.1 Site Area Emergency

Loss or potential loss of **any** two barriers (Table F-1, Attachment 9.6)

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

None

PNPS Basis:

Fuel clad, RCS, and Primary Containment comprise the fission product barriers. Table F-1 (Attachment 9.6) lists the fission product barrier thresholds, bases, and references.

At the Site Area Emergency classification level, each barrier is weighted equally. A Site Area Emergency is therefore appropriate for any combination of the following conditions:

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

Site Area Emergency - FS1.1

At the Site Area Emergency classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of fuel clad and RCS barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Primary Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both fuel clad and RCS potential loss thresholds existed, the Emergency Director would have greater assurance that escalation to a General Emergency is less imminent.

PNPS Basis Reference(s):

None

General Emergency - FG1.1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss of **any** two barriers and loss or potential loss of third barrier

EAL:

FG1.1 General Emergency

Loss of **any** two barriers

AND

Loss or potential loss of third barrier (Table F-1, Attachment 9.6)

Mode Applicability:

1 - Run, 2 - Startup, 3 - Hot Shutdown

NEI 99-01 Basis:

None

PNPS Basis:

Fuel clad, RCS, and Primary Containment comprise the fission product barriers. Table F-1 (Attachment 9.6) lists the fission product barrier thresholds, bases, and references.

At the General Emergency classification level each barrier is weighted equally. A General Emergency is therefore appropriate for any combination of the following conditions:

- Loss of fuel clad, RCS, and Primary Containment barriers
- Loss of fuel clad and RCS barriers with potential loss of Primary Containment barrier
- Loss of RCS and Primary Containment barriers with potential loss of fuel clad barrier
- Loss of fuel clad and Primary Containment barriers with potential loss of RCS barrier

General Emergency - FG1.1

PNPS Basis Reference(s):

None

Category C - Cold Shutdown/Refueling System Malfunction

EAL Group: Cold Conditions (RCS temperature ≤ 212°F); EALs in this category are applicable only in one or more cold operating modes.

Category C EALs are directly associated with cold shutdown or refueling system safety functions. Given the variability of plant configurations (e.g., systems out of service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The cold shutdown and refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, containment closure, and fuel clad integrity for the applicable operating modes (4 - Cold Shutdown, 5 - Refuel, D - Defueled).

The events of this category pertain to the following subcategories:

1. Loss of AC Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite sources for 4160V emergency buses.

2. RPV Level

RPV water level is a measure of inventory available to ensure adequate core cooling and, therefore, maintain fuel clad integrity. The RPV provides a volume for the coolant that covers the reactor core. The RPV and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail.

3. RCS Temperature

Uncontrolled or inadvertent temperature or pressure increases are indicative of a potential loss of safety functions.

4. Communications

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

5. Inadvertent Criticality

Inadvertent criticalities pose potential personnel safety hazards as well being indicative of losses of reactivity control.

6. Loss of DC Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of vital 125 volt DC power sources.

Unusual Event - CU1.1

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 1 - Loss of Power

Initiating Condition: AC power capability to emergency buses reduced to a single power source for 15 minutes or longer such that **any** additional single failure would result in loss of **all** AC power to emergency buses

EAL:

CU1.1 Unusual Event

AC power capability to emergency buses A5 and A6 reduced to a single power source (Table C-4) for ≥ 15 min. such that **any** additional single failure would result in loss of **all** AC power to emergency buses (Note 3)

Note 3: The Emergency Director should **not** wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table C-4 AC Power Sources
Offsite
- Startup Transformer (X4) - Shutdown Transformer - UAT - Backscuttle via Main Transformer (only if already established)
Onsite
- EDG A - EDG B - SBO DG

Unusual Event - CU1.1

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

NEI 99-01 Basis:

The condition indicated by this EAL is the degradation of the offsite and onsite AC 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 all but one emergency generator to supply power to its emergency buses. The subsequent loss of this single power source would escalate the event to an Alert in accordance with EAL CA1.1.

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

PNPS Basis:

The PNPS emergency 4160V AC electrical distribution system is illustrated in Figure C-1.

4160V AC buses A5 and A6 are the emergency buses. These electrical buses provide power to vital equipment such as RHR, Core Spray, and CRD pumps as well as vital 480V AC transformers feeding buses B1 and B2. Offsite power supply transformers are those transformers which are capable of providing power to the emergency buses independent of the onsite generators. This EAL does consider backscuttle capability. But due to the significant time required to establish the necessary switching lineups and removal of generator disconnect links, backscuttle is not considered an available offsite power supply transformer unless already established at the time of the loss of other offsite power supply transformers.

This cold condition EAL is equivalent to the hot condition loss of all offsite AC power EAL SA1.1.

Unusual Event - CU1.1

PNPS Basis Reference(s):

1. Bechtel Drawing Electrical Single Line Diagram S-E-155
2. FSAR Section 8.1
3. FSAR Section 8.3
4. FSAR Section 8.5
5. PNPS 5.3.31, *"Station Blackout"*
6. PNPS 2.2.8, *"Standby AC Power System (Diesel Generators)"*
7. PNPS 2.2.146, *"Station Blackout Diesel Generator"*
8. PNPS 2.4.144, *"Degraded Voltage"*
9. PNPS 2.4.A.23, *"Loss/Degradation Of 23kV Line"*
10. PNPS 2.4.A.5, *"Loss of Electrical Bus A5"*
11. PNPS 2.4.A.6, *"Loss of Electrical Bus A6"*

ATTACHMENT 9.5 **CATEGORY C, COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION**

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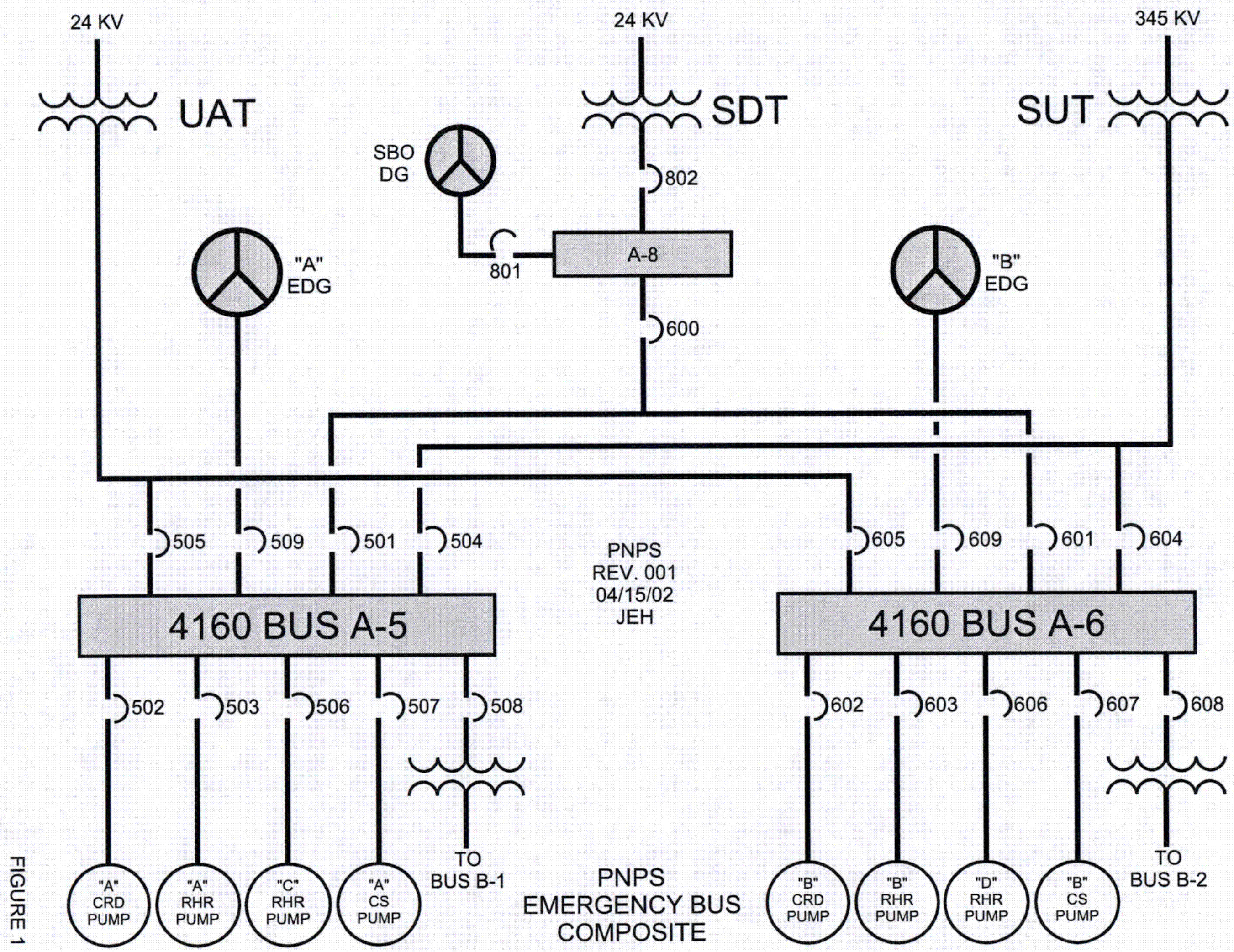
Unusual Event - CU1.1

Figure C-1 Emergency Bus Composite Diagram



Alert - CA1.1

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 1 - Loss of Power

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

EAL:

CA1.1 Alert

Loss of **all** offsite and **all** onsite AC power (Table C-4) to emergency buses A5 and A6 for ≥ 15 min. (Note 3)

Note 3: The Emergency Director should **not** wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table C-4 AC Power Sources
Offsite
- Startup Transformer (X4) - Shutdown Transformer - UAT - Backscuttle via Main Transformer (only if already established)
Onsite
- EDG A - EDG B - SBO DG

Alert - CA1.1

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel, D - Defueled

NEI 99-01 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.

The event can be classified as an Alert when in cold shutdown, refueling, or defueled mode because of the significantly reduced decay heat and lower temperature and pressure, increasing the time to restore one of the emergency buses, relative to that specified for the Site Area Emergency EAL.

Escalating to Site Area Emergency, if appropriate, is by Category A EALs.

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

PNPS Basis:

This EAL is indicated by the loss of all offsite and onsite AC power to the emergency buses A5 and A6.

The PNPS emergency 4160V AC electrical distribution system is illustrated in Figure C-1.

The inability to provide emergency AC power from any offsite power source poses a potential degradation of reactor plant safety. If onsite power capability becomes degraded, significant losses of vital equipment operability may occur. 4160V AC buses A5 and A6 are the emergency buses. These electrical buses provide power to vital equipment such as RHR, Core Spray, and CRD pumps as well as vital 480V AC transformers feeding buses B1 and B2. Offsite power supply transformers are those transformers which are capable of providing power to the emergency buses independent of the onsite generators. This EAL does consider backscuttle capability. But due to the significant time required to establish the necessary switching lineups and removal of generator disconnect links, backscuttle is not considered an available offsite power supply transformer unless already established at the time of the loss of other offsite power supply transformers.

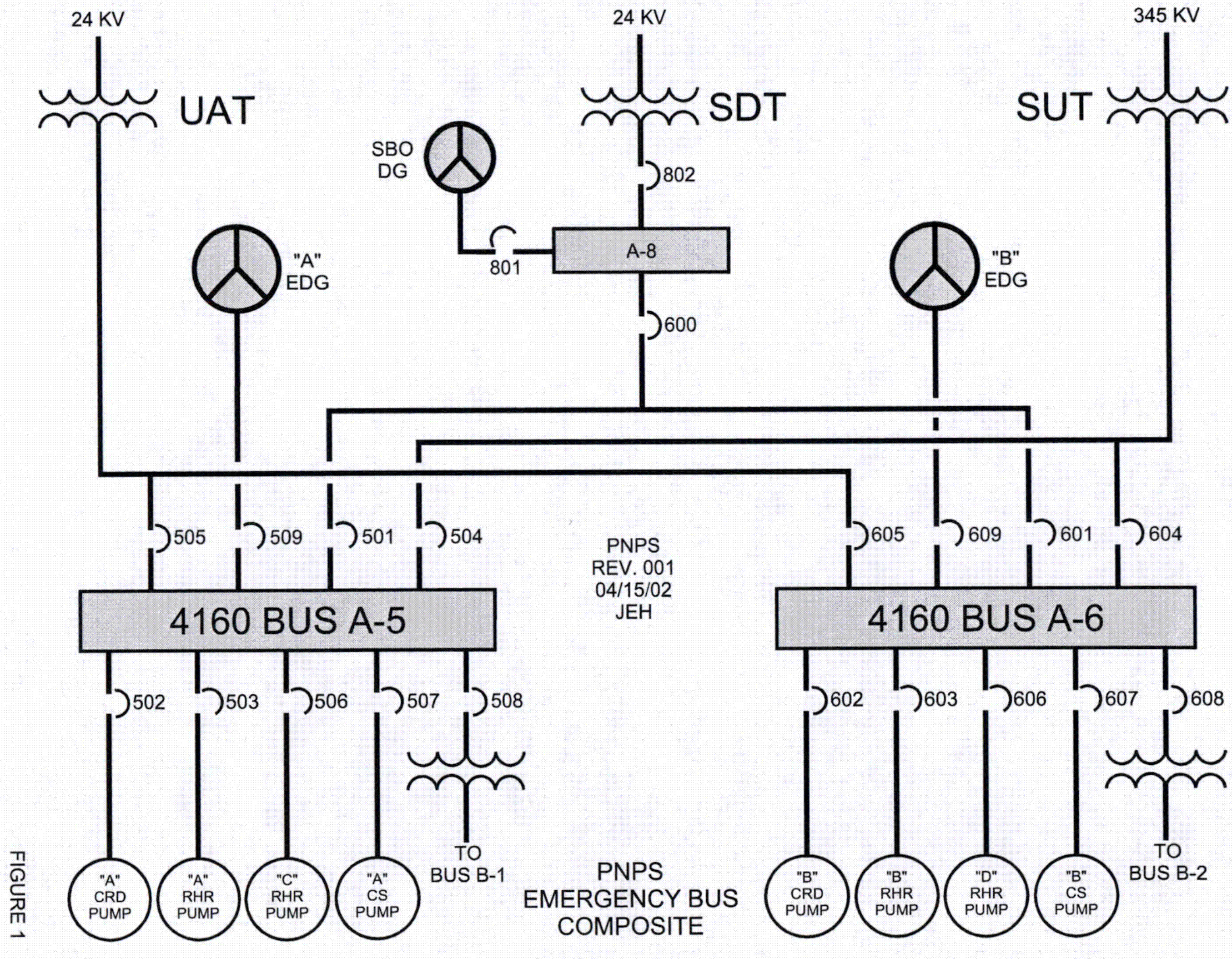
This EAL is the cold condition equivalent of the hot condition loss of all AC power EAL SS1.1.

PNPS Basis Reference(s):

1. Bechtel Drawing Electrical Single Line Diagram S-E-155
2. FSAR Section 8.1
3. FSAR Section 8.3
4. FSAR Section 8.5
5. PNPS 5.3.31, *"Station Blackout"*
6. PNPS 2.2.8, *"Standby AC Power System (Diesel Generators)"*
7. PNPS 2.2.146, *"Station Blackout Diesel Generator"*
8. PNPS 2.4.144, *"Degraded Voltage"*
9. PNPS 2.4.A.23, *"Loss/Degradation Of 23kV Line"*
10. PNPS 2.4.A.5, *"Loss of Electrical Bus A5"*
11. PNPS 2.4.A.6, *"Loss of Electrical Bus A6"*

Alert - CA1.1

Figure C-1 Emergency Bus Composite Diagram



Unusual Event - CU2.1

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 2 - RPV Level

Initiating Condition: RCS leakage

EAL:

CU2.1	**Unusual Event**
RPV level **cannot** be restored and maintained > +12 in. for ≥ 15 min. (Note 3)	

Note 3: The Emergency Director should **not** wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

4 - Cold Shutdown

NEI 99-01 Basis:

This EAL is considered to be a potential degradation of the level of safety of the plant. The inability to maintain or restore level is indicative of loss of RCS inventory.

Relief valve normal operation should be excluded from this EAL. However, a relief valve that operates and fails to close according to design should be considered applicable to this EAL if the relief valve cannot be isolated.

Prolonged loss of RCS Inventory may result in escalation to the Alert emergency classification level via either CA2.1 or CA3.1.

PNPS Basis:

The condition of this EAL may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. When RPV level drops to 12 in. (low level scram setpoint), level is well below the normal control band and automatic RPS and PCIS actuations are required (ref. 1).

All Control Room RPV water level monitors have their indicated range referenced to an elevation of 482.5 inches above the bottom of the reactor vessel, 126.3 inches above the top of the active fuel (TAF). The common zero reference for RPV water level instrumentation was chosen as the bottom of the steam separators, 482.5 inches above the bottom of the reactor vessel. The low level instruments cover the level from 142" above vessel zero to 542" above vessel zero. The wide range instrument covers the upper 400" of the vessel from 425" above vessel zero to top of the vessel. Level Indicators LI-263-106A & B, LI-263-73A & B, LI-1001-650A & B, and Level Recorders LR-1001-604A & B (Pen 3) are used to monitor RPV level during accident conditions. Level Recorder LR-640-28 is used to monitor RPV level during shutdown conditions (ref. 2).

In preparation for refueling operations, a temporary shutdown/floodup transmitter is installed on Reactor Building 51' Rack C2206 and the narrow range feedwater level indicator LI-640-29B is replaced with a reactor shutdown/floodup digital level display with visual alarm points (EPIC point RXX054). This allows monitoring RPV level between "normal range" (instrument 0 to 466") through flood elevation (117') (ref. 2).

Figure C-3 illustrates the elevations of the temporary RPV level floodup instrument (ref. 3).

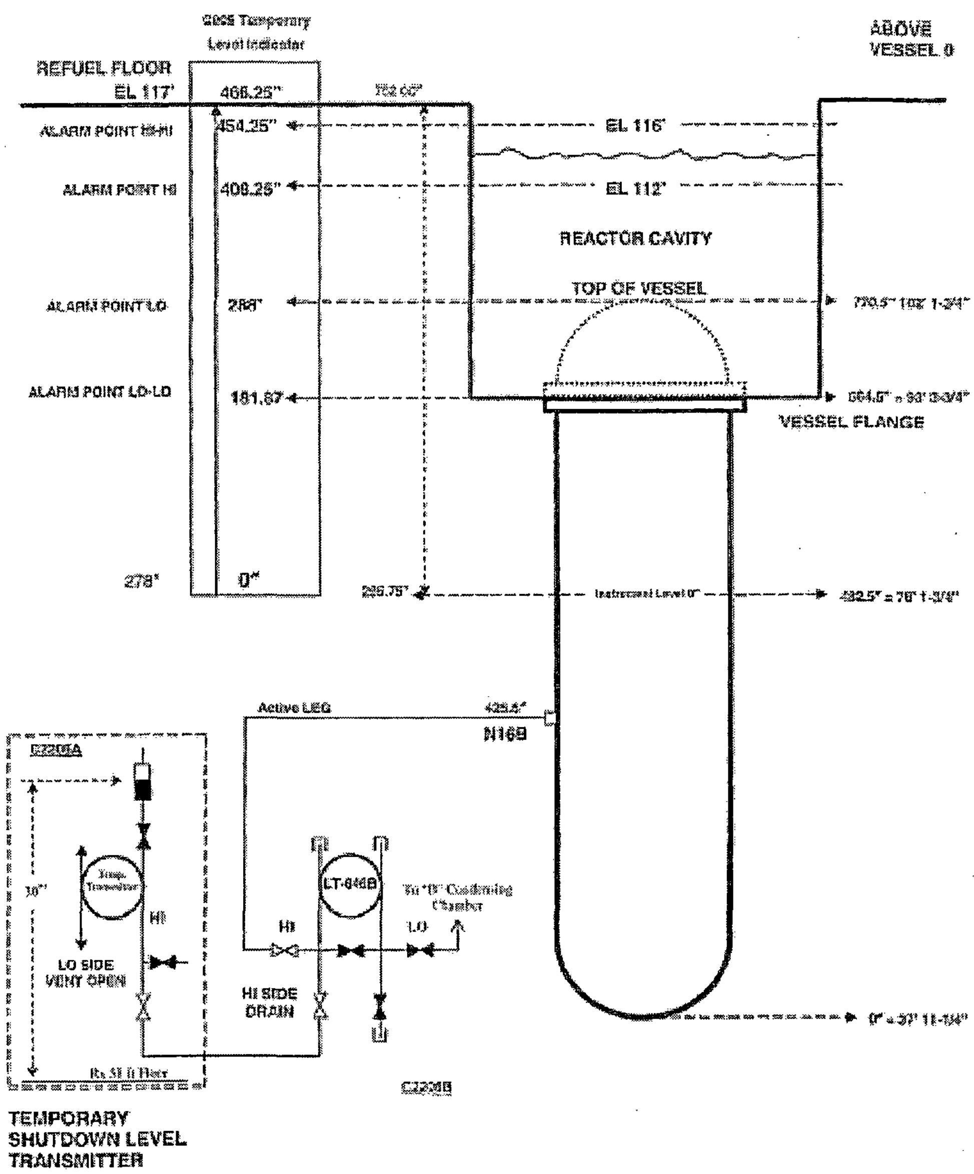
This Cold Shutdown EAL represents the hot condition EAL SU6.1 in which RCS leakage is associated with Technical Specifications limits. In Cold Shutdown, these limits are not applicable; hence, the use of RPV level as the parameter of concern in this EAL (ref. 4).

PNPS Basis Reference(s):

1. EOP-1, *"RPV Control"*
2. PNPS 2.2.80, *"Reactor Vessel Level, Temperature, And Internal Pressure Instrumentation"*
3. PNPS 3.M.2-40, *"Refuel Outage Temporary Modification Reactor Shutdown/Flood-Up Level Indication"*
4. NEI/NRC EAL FAQ #2006-014

Unusual Event - CU2.1

Figure C-3 Floodup RPV Level Indication



Unusual Event - CU2.2

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 2 - RPV Level

Initiating Condition: Unplanned loss of RPV inventory

EAL:

CU2.2 Unusual Event

Unplanned RPV level drop for ≥ 15 min (Note 3) below **EITHER**:

RPV flange (+182 in.)

OR

RPV level band when the RPV level band is established below the RPV flange

Note 3: The Emergency Director should **not** wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

5 - Refuel

NEI 99-01 Basis:

This EAL is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RPV water level below the RPV flange are carefully planned and procedurally controlled. An unplanned event that results in water level decreasing below the RPV flange, or below the planned RPV water level for the given evolution (if the planned RPV water level is already below the RPV flange), warrants declaration of a UE due to the reduced RPV inventory that is available to keep the core covered.

The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame, then it may indicate a more serious condition exists.

Continued loss of RCS Inventory will result in escalation to the Alert emergency classification level via either EAL CA2.1 or EAL CA3.1.

This EAL involves a decrease in RCS level below the top of the RPV flange that continues for 15 minutes due to an unplanned event. This EAL is not applicable to decreases in flooded reactor cavity level, which is addressed by EAL AU2.1, until such time as the level decreases to the level of the vessel flange.

PNPS Basis:

181.87 in. (rounded to 182 in. for readability) on the Shutdown RPV water level instrument corresponds to the RPV flange (ref. 1).

All Control Room RPV water level monitors have their indicated range referenced to an elevation of 482.5 inches above the bottom of the reactor vessel, 126.3 inches above the top of the active fuel (TAF). The common zero reference for RPV water level instrumentation was chosen as the bottom of the steam separators, 482.5 inches above the bottom of the reactor vessel. The low level instruments cover the level from 142" above vessel zero to 542" above vessel zero. The wide range instrument covers the upper 400" of the vessel from 425" above vessel zero to top of the vessel. Level Indicators LI-263-106A & B, LI-263-73A & B, LI-1001-650A & B, and Level Recorders LR-1001-604A & B (Pen 3) are used to monitor RPV level during accident conditions. Level Recorder LR-640-28 is used to monitor RPV level during shutdown conditions (ref. 2).

In preparation for refueling operations, a temporary shutdown/floodup transmitter is installed on Reactor Building 51' Rack C2206 and the narrow range feedwater level indicator LI-640-29B is replaced with a reactor shutdown/floodup digital level display with visual alarm points (EPIC point RXX054). This allows monitoring RPV level between "normal range" (instrument 0 to 466") through flood elevation (117') (ref. 1, 2).

Figure C-3 illustrates the elevations of the temporary RPV level floodup instrument (ref. 1).

Unusual Event - CU2.2

PNPS Basis Reference(s):

1. PNPS 3.M.2-40, *"Refuel Outage Temporary Modification Reactor Shutdown/Flood-Up Level Indication"*
2. PNPS 2.2.80, *"Reactor Vessel Level, Temperature, And Internal Pressure Instrumentation"*

ATTACHMENT 9.5 **CATEGORY C, COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION**

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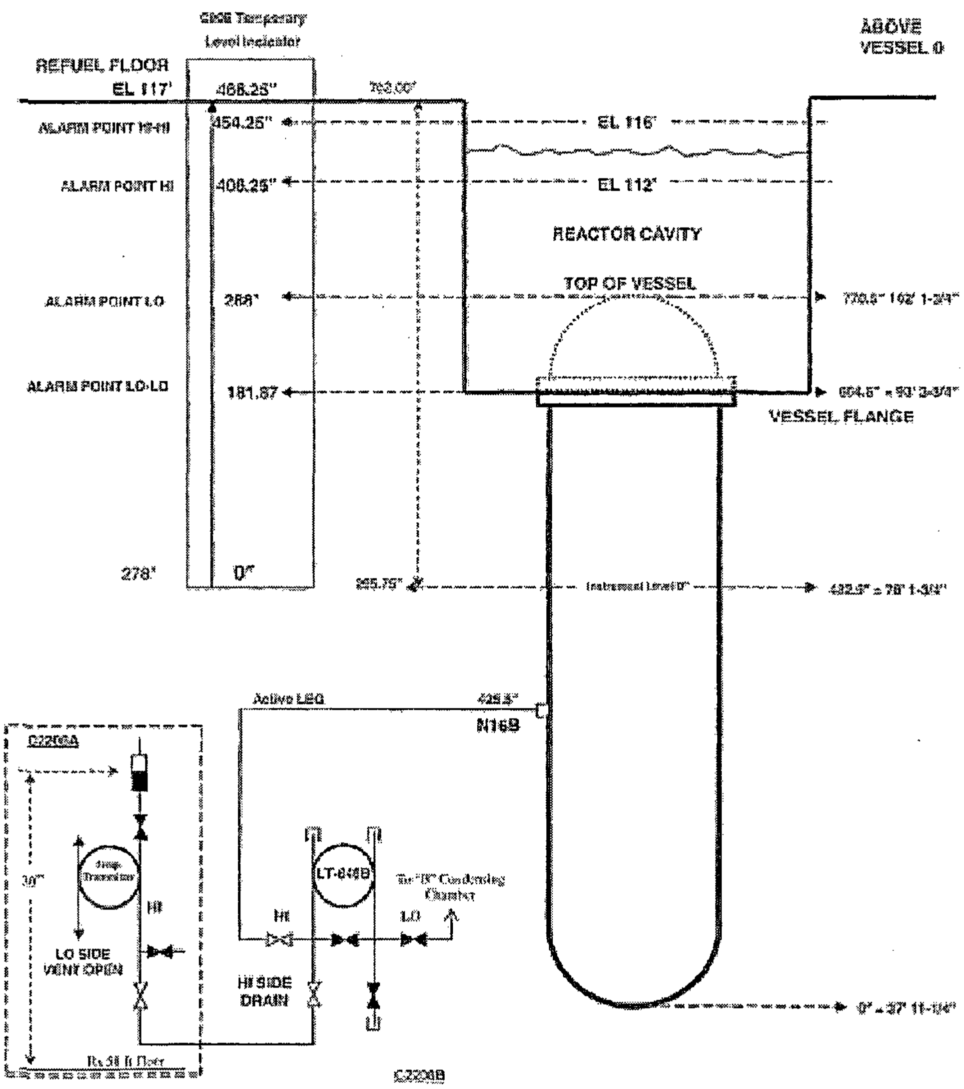
Unusual Event - CU2.2

Figure C-3 Floodup RPV Level Indication



Unusual Event - CU2.3

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 2 - RPV Level

Initiating Condition: Unplanned loss of RPV inventory

EAL:

CU2.3 Unusual Event

RPV level **cannot** be monitored with **any** unexplained RPV leakage indication, Table C-1

Table C-1 RPV Leakage Indications
- Drywell equipment drain sump level rise
- Drywell floor drain sump level rise
- Reactor Building equipment drain sump level rise
- Reactor Building floor drain sump level rise
- Torus level rise
- RPV make-up rate rise
- Observation of unisolable RCS leakage

Mode Applicability:

5 - Refuel

NEI 99-01 Basis:

This EAL is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RPV water level below the RPV flange are carefully planned and procedurally controlled. An unplanned event that results in water level decreasing below the RPV flange, or below the planned RPV water level for the given evolution (if the planned RPV water level is already below the RPV flange), warrants declaration of a UE due to the reduced RPV inventory that is available to keep the core covered.

Continued loss of RCS inventory will result in escalation to the Alert emergency classification level via either EAL CA2.1 or EAL CA3.1.

This EAL addresses conditions in the refueling mode when normal means of core temperature indication and RPV level indication may not be available. Redundant means of RPV 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 RPV 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.

PNPS Basis:

All Control Room RPV water level monitors have their indicated range referenced to an elevation of 482.5 inches above the bottom of the reactor vessel, 126.3 inches above the top of the active fuel (TAF). The common zero reference for RPV water level instrumentation was chosen as the bottom of the steam separators, 482.5 inches above the bottom of the reactor vessel. The low level instruments cover the level from 142" above vessel zero to 542" above vessel zero. The wide range instrument covers the upper 400" of the vessel from 425" above vessel zero to top of the vessel. Level Indicators LI-263-106A & B, LI-263-73A & B, LI-1001-650A & B, and Level Recorders LR-1001-604A & B (Pen 3) are used to monitor RPV level during accident conditions. Level Recorder LR-640-28 is used to monitor RPV level during shutdown conditions (ref. 2).

Unusual Event - CU2.3

In preparation for refueling operations, a temporary shutdown/floodup transmitter is installed on Reactor Building 51' Rack C2206 and the narrow range feedwater level indicator LI-640-29B is replaced with a reactor shutdown/floodup digital level display with visual alarm points (EPIC point RXX054). This allows monitoring RPV level between "normal range" (instrument 0 to 466") through flood elevation (117') (ref. 1, 2).

Figure C-3 illustrates the elevations of the temporary RPV level floodup instrument (ref. 1).

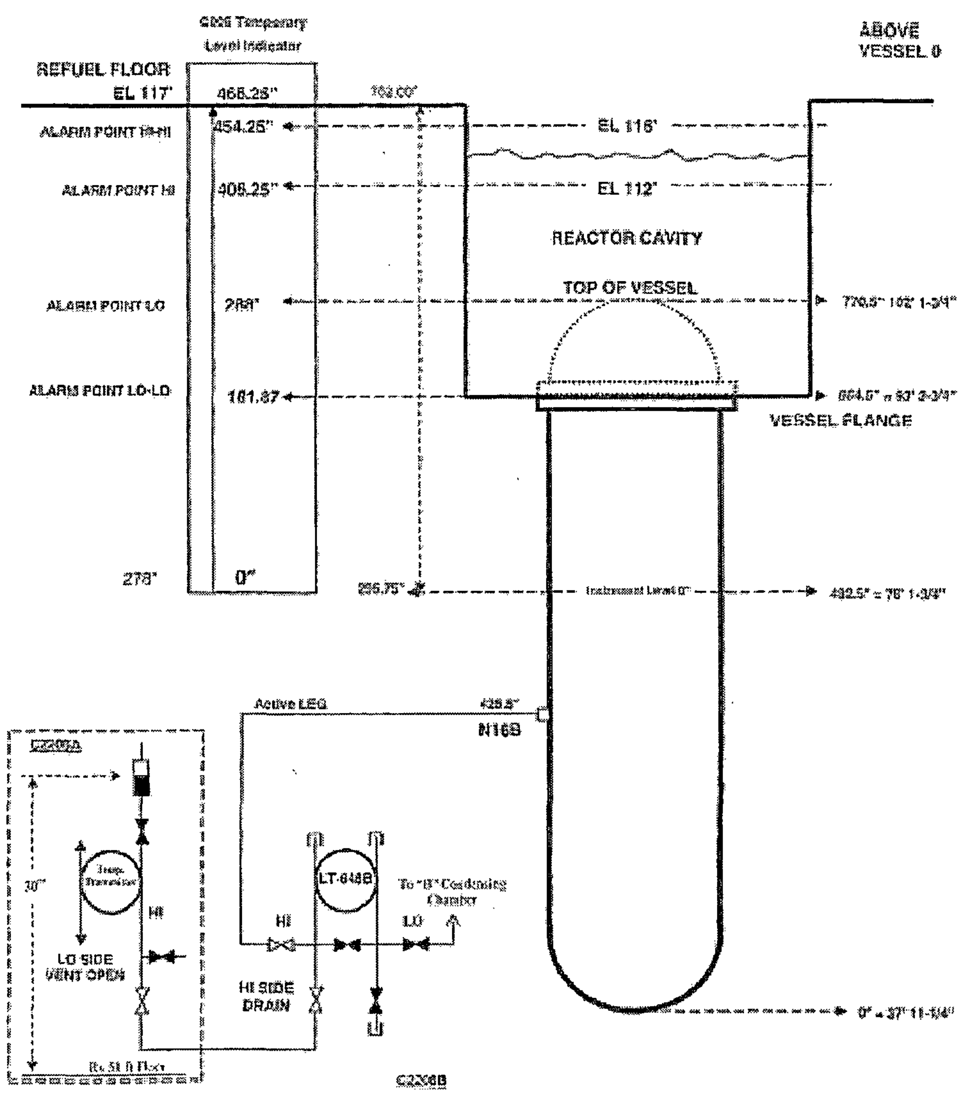
In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV. A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus level could be indicative of RHR valve misalignment or leakage. If the makeup rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory (ref. 3, 4, 5, 6, 7).

PNPS Basis Reference(s):

1. PNPS 3.M.2-40, *"Refuel Outage Temporary Modification Reactor Shutdown/Flood-Up Level Indication"*
2. PNPS 2.2.80, *"Reactor Vessel Level, Temperature, And Internal Pressure Instrumentation"*
3. PNPS 2.2.77, *"Drywell Leak Detection Systems"*
4. PNPS 2.5.2.71, *"Radwaste Collection System"*
5. FSAR Section 4.10 - Nuclear System Leakage Rate Limits
6. FSAR Section 5.2 - Primary Containment System
7. FSAR Section 9.2 - Liquid Radwaste System

Unusual Event - CU2.3

Figure C-3 Floodup RPV Level Indication



Alert - CA2.1

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 2 - RPV Level

Initiating Condition: Loss of RPV inventory

EAL:

CA2.1 Alert

RPV level < -45 in.

OR

RPV level **cannot** be monitored for ≥ 15 min. with **any** unexplained RPV leakage indication, Table C-1 (Note 3)

Note 3: The Emergency Director should **not** wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table C-1 RPV Leakage Indications
- Drywell equipment drain sump level rise
- Drywell floor drain sump level rise
- Reactor Building equipment drain sump level rise
- Reactor Building floor drain sump level rise
- Torus level rise
- RPV make-up rate rise
- Observation of unisolable RCS leakage

Alert - CA2.1

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

NEI 99-01 Basis:

This EAL serves as a precursor to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV level decrease and potential core uncovery. This condition will result in a minimum emergency classification level of an Alert.

The inability to restore and maintain level after reaching this setpoint (-45 in.) would be indicative of a failure of the RCS barrier.

If RPV level continues to lower, escalation to Site Area Emergency will be via EAL CS2.1 or CS2.2.

PNPS Basis:

The threshold RPV level of -45 in. is the low-low ECCS actuation setpoint (ref. 1). Low-low level switches LIS-263-72A, B, C, and D initiate Reactor Core Injection Cooling (RCIC), Automatic Depressurization System (ADS), and start the standby diesel generator (DG) at -46.3 in. (rounded to -45 in. for readability in accordance with EOPs). Core Spray (CS) is also initiated if Reactor pressure is less than 395 to 405 psig or after a 9 to 15.4-minute time delay.

Low level switches LS-263-72A-1, B-1, C-1, and D-1 initiate High Pressure Core Injection (HPCI) and start the standby DG at this level. Residual Heat Removal (RHR) will also be initiated (ref. 3).

Alert - CA2.1

All Control Room RPV water level monitors have their indicated range referenced to an elevation of 482.5 inches above the bottom of the reactor vessel, 126.3 inches above the top of the active fuel (TAF). The common zero reference for RPV water level instrumentation was chosen as the bottom of the steam separators, 482.5 inches above the bottom of the reactor vessel. The low level instruments cover the level from 142" above vessel zero to 542" above vessel zero. The wide range instrument covers the upper 400" of the vessel from 425" above vessel zero to top of the vessel. Level Indicators LI-263-106A & B, LI-263-73A & B, LI-1001-650A & B, and Level Recorders LR-1001-604A & B (Pen 3) are used to monitor RPV level during accident conditions. Level Recorder LR-640-28 is used to monitor RPV level during shutdown conditions (ref. 3).

In preparation for refueling operations, a temporary shutdown/floodup transmitter is installed on Reactor Building 51' Rack C2206 and the narrow range feedwater level indicator LI-640-29B is replaced with a reactor shutdown/floodup digital level display with visual alarm points (EPIC point RXX054). This allows monitoring RPV level between "normal range" (instrument 0 to 466") through flood elevation (117') (ref. 2, 3).

Figure C-3 illustrates the elevations of the temporary RPV level floodup instrument (ref. 2).

Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV. A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus level could be indicative of RHR valve misalignment or leakage. If the makeup rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory (ref. 4, 5, 6, 7, 8).

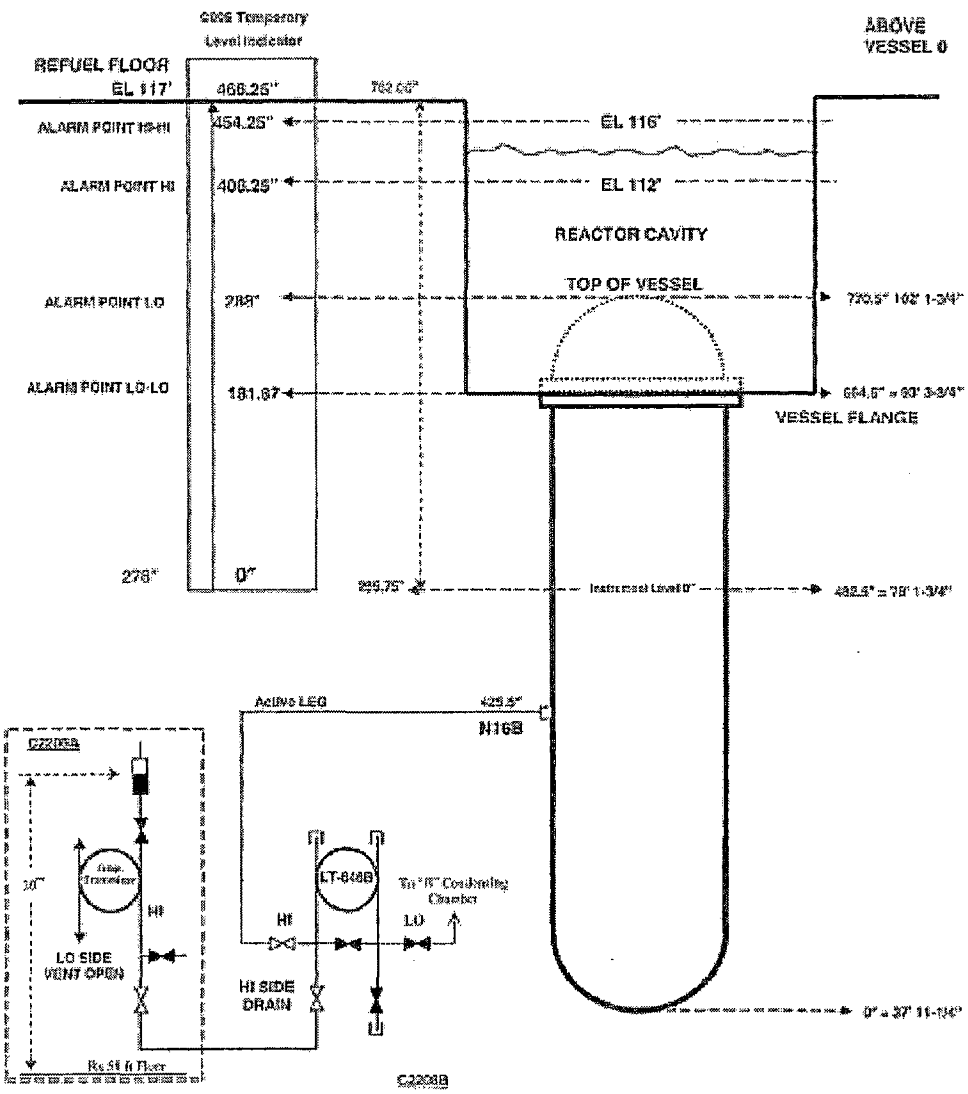
Alert - CA2.1

PNPS Basis Reference(s):

1. EOP-1, *"RPV Control"*
2. PNPS 3.M.2-40, *"Refuel Outage Temporary Modification Reactor Shutdown/Flood-Up Level Indication"*
3. PNPS 2.2.80, *"Reactor Vessel Level, Temperature, and Internal Pressure Instrumentation"*
4. PNPS 2.2.77, *"Drywell Leak Detection Systems"*
5. PNPS 2.5.2.71, *"Radwaste Collection System"*
6. FSAR Section 4.10 - Nuclear System Leakage Rate Limits
7. FSAR Section 5.2 - Primary Containment System
8. FSAR Section 9.2 - Liquid Radwaste System

Alert - CA2.1

Figure C-3 Floodup RPV Level Indication



Procedure/Document Number: EP-AD-601	Revision: 6
Equipment/Facility/Other: Pilgrim Nuclear Power Station	
Title: Emergency Action Level Technical Bases Document	

Part V. Emergency Planning Element/Function Screen (Associated 10 CFR 50.47(b) planning standard function identified in brackets) Does this activity affect any of the following, including program elements from NUREG-0654/FEMA REP-1 Section II?	
1. Responsibility for emergency response is assigned. [1]	☐
2. The response organization has the staff to respond and to augment staff on a continuing basis (24/7 staffing) in accordance with the emergency plan. [1]	☐
3. The process ensures that on shift emergency response responsibilities are staffed and assigned. [2]	☐
4. The process for timely augmentation of onshift staff is established and maintained. [2]	☐
5. Arrangements for requesting and using off site assistance have been made. [3]	☐
6. State and local staff can be accommodated at the EOF in accordance with the emergency plan. [3]	☐
7. A standard scheme of emergency classification and action levels is in use. [4]	☐
8. Procedures for notification of State and local governmental agencies are capable of alerting them of the declared emergency within 15 minutes after declaration of an emergency and providing follow-up notifications. [5]	☐
9. Administrative and physical means have been established for alerting and providing prompt instructions to the public within the plume exposure pathway. [5]	☐
10. The public ANS meets the design requirements of FEMA-REP-10, Guide for Evaluation of Alert and Notification Systems for Nuclear Power Plants, or complies with the licensee's FEMA-approved ANS design report and supporting FEMA approval letter. [5]	☐
11. Systems are established for prompt communication among principal emergency response organizations. [6]	☐
12. Systems are established for prompt communication to emergency response personnel. [6]	☐
13. Emergency preparedness information is made available to the public on a periodic basis within the plume exposure pathway emergency planning zone (EPZ). [7]	☐
14. Coordinated dissemination of public information during emergencies is established. [7]	☐
15. Adequate facilities are maintained to support emergency response. [8]	☐
16. Adequate equipment is maintained to support emergency response. [8]	☐
17. Methods, systems, and equipment for assessment of radioactive releases are in use. [9]	☐
18. A range of public PARs is available for implementation during emergencies. [10]	☐
19. Evacuation time estimates for the population located in the plume exposure pathway EPZ are available to support the formulation of PARs and have been provided to State and local governmental authorities. [10]	☐
20. A range of protective actions is available for plant emergency workers during emergencies, including those for hostile action events.[10]	☐

Site Area Emergency - CS2.1

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 2 - RPV Level

Initiating Condition: Loss of RPV inventory affecting core decay heat removal capability

EAL:

CS2.1 Site Area Emergency

With Containment Closure **not** established, RPV level < -50 in. (Note 4)

Note 4: Containment closure is the action taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment closure is established when Primary or Secondary Containment integrity is established in accordance with Section 3.7 of Technical Specifications.

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

NEI 99-01 Basis:

Under the conditions specified by this EAL, continued decrease in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via EAL CG2.1, CG2.2, AG1.1, AG1.2, or AG1.3.

Site Area Emergency - CS2.1

PNPS Basis:

When RPV level decreases to -50 in. on the normal range level indicators (i.e., the -50 to +50 instruments), water level is 5 inches below the low-low ECCS actuation setpoint (ref. 1, 4).

The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV level decrease and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and potential loss of the fuel clad barrier.

Containment closure is the action taken to secure either Primary Containment or Secondary Containment and the associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions (ref. 2). As applied to PNPS, containment closure is established when either Primary Containment integrity or Secondary Containment integrity is established in accordance with Section 3.7 of Technical Specifications (ref. 3).

PNPS Basis Reference(s):

1. EOP-1, *"RPV Control"*
2. NEI 99-01 Definitions and Appendix C
3. Technical Specifications Section 3.7
4. PNPS 2.2.80, *"Reactor Vessel Level, Temperature, and Internal Pressure Instrumentation"*

Site Area Emergency - CS2.2

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 2 - RPV Level

Initiating Condition: Loss of RPV inventory affecting core decay heat removal capability

EAL:

CS2.2 Site Area Emergency

With containment closure established, RPV level < -125 in. (Note 4)

Note 4: Containment closure is the action taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment closure is established when Primary or Secondary Containment integrity is established in accordance with Section 3.7 of Technical Specifications.

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

NEI 99-01 Basis:

Under the conditions specified by this EAL, continued decrease in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via CG2.1, CG2.2, AG1.1, AG1.2, or AG1.3.

Site Area Emergency - CS2.2

PNPS Basis:

When RPV level drops below -125 in., core uncovery is about to occur (ref. 1).

The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV level decrease and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and potential loss of the fuel clad barrier.

Containment closure is the action taken to secure either Primary Containment or Secondary Containment and the associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions (ref. 2). As applied to PNPS, containment closure is established when either Primary Containment integrity or Secondary Containment integrity is established in accordance with Section 3.7 of Technical Specifications (ref. 3).

PNPS Basis Reference(s):

1. EOP-1, "*RPV Control*"
2. NEI 99-01 Definitions and Appendix C
3. Technical Specifications Section 3.7

Site Area Emergency - CS2.3

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 2 - RPV Level

Initiating Condition: Loss of RPV inventory affecting core decay heat removal capability

EAL:

CS2.3 Site Area Emergency

RPV level **cannot** be monitored for ≥ 30 min. (Note 3) with a loss of inventory as indicated by **EITHER**:

Unexplained RPV leakage indication, Table C-1

OR

Erratic Source Range Monitor indication

Note 3: The Emergency Director should **not** wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table C-1 RPV Leakage Indications
- Drywell equipment drain sump level rise
- Drywell floor drain sump level rise
- Reactor Building equipment drain sump level rise
- Reactor Building floor drain sump level rise
- Torus level rise
- RPV make-up rate rise
- Observation of unisolable RCS leakage

Site Area Emergency - CS2.3

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

NEI 99-01 Basis:

Under the conditions specified by this EAL, continued decrease in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via EAL CG2.1, CG2.2, AG1.1, AG1.2, or AG1.3.

The 30-minute duration allows sufficient time for actions to be performed to recover inventory control equipment.

PNPS Basis:

All Control Room RPV water level monitors have their indicated range referenced to an elevation of 482.5 inches above the bottom of the reactor vessel, 126.3 inches above the top of the active fuel (TAF). The common zero reference for RPV water level instrumentation was chosen as the bottom of the steam separators, 482.5 inches above the bottom of the reactor vessel. The low level instruments cover the level from 142" above vessel zero to 542" above vessel zero. The wide range instrument covers the upper 400" of the vessel from 425" above vessel zero to top of the vessel. Level Indicators LI-263-106A & B, LI-263-73A & B, LI-1001-650A & B, and Level Recorders LR-1001-604A & B (Pen 3) are used to monitor RPV level during accident conditions. Level Recorder LR-640-28 is used to monitor RPV level during shutdown conditions (ref. 2).

In preparation for refueling operations, a temporary shutdown/floodup transmitter is installed on Reactor Building 51' Rack C2206 and the narrow range feedwater level indicator LI-640-29B is replaced with a reactor shutdown/floodup digital level display with visual alarm points (EPIC point RXX054). This allows monitoring RPV level between "normal range" (instrument 0 to 466") through flood elevation (117') (ref. 1, 2).

Figure C-3 illustrates the elevations of the temporary RPV level floodup instrument (ref. 1).

Site Area Emergency - CS2.3

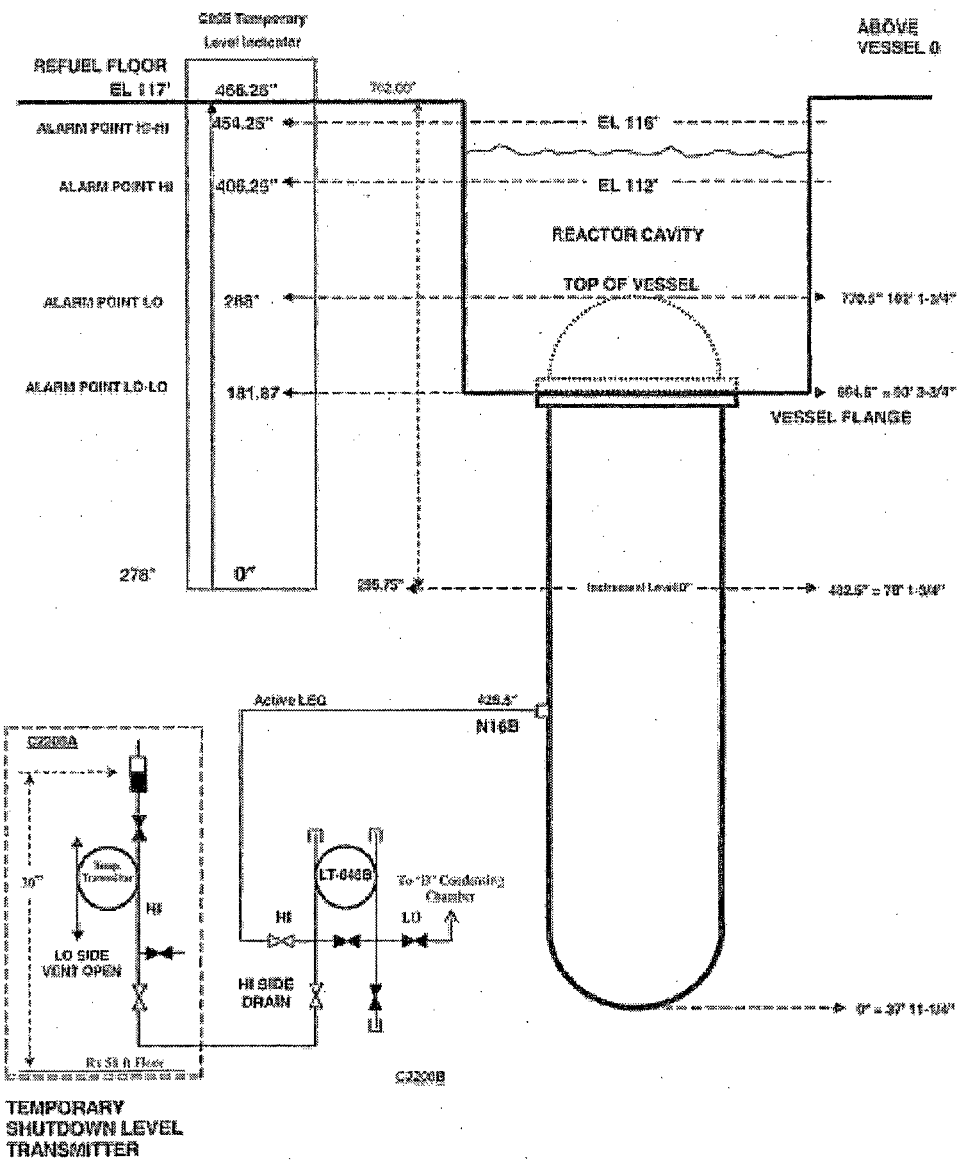
In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV. A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus level could be indicative of RHR valve misalignment or leakage. If the makeup rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory (ref. 3, 4, 5, 6, 7).

PNPS Basis Reference(s):

1. PNPS 3.M.2-40, *"Refuel Outage Temporary Modification Reactor Shutdown/Flood-Up Level Indication"*
2. PNPS 2.2.80, *"Reactor Vessel Level, Temperature, and Internal Pressure Instrumentation"*
3. PNPS 2.2.77, *"Drywell Leak Detection Systems"*
4. PNPS 2.5.2.71, *"Radwaste Collection System"*
5. FSAR Section 4.10 - Nuclear System Leakage Rate Limits
6. FSAR Section 5.2 - Primary Containment System
7. FSAR Section 9.2 - Liquid Radwaste System

Site Area Emergency - CS2.3

Figure C-3 Floodup RPV Level Indication



General Emergency - CG2.1

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 2 - RPV Level

Initiating Condition: Loss of RPV inventory affecting fuel clad integrity with containment challenged

EAL:

CG2.1 General Emergency

RPV level < -125 in. for ≥ 30 min. (Note 3)

AND

Any containment challenge indication, Table C-5

Note 3: The Emergency Director should not wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Table C-5 Containment Challenge Indications
- Containment Closure **not** established (Note 4)
- Deflagration concentrations exist inside PC
- Unplanned rise in PC pressure
- Secondary Containment area radiation > **any** Maximum Safe Operating Value (EOP-4, Table L)

Note 4: Containment closure is the action taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment closure is established when Primary or Secondary Containment integrity is established in accordance with Section 3.7 of Technical Specifications.

General Emergency - CG2.1

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

NEI 99-01 Basis:

This EAL represents the inability to restore and maintain RPV level to above the top of active fuel with containment challenged. Fuel damage is probable if RPV level cannot be restored as available decay heat will cause boiling, further reducing the RPV level. With the containment breached or challenged, 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. The GE is declared on the occurrence of the loss or imminent loss of function of all three barriers.

A number of variables can have a significant impact on heat removal capability challenging the fuel clad barrier. Examples include initial vessel level and shutdown heat removal system design.

Analysis indicates that core damage may occur within an hour following continued core uncovery; therefore, 30 minutes was conservatively chosen.

If containment closure is re-established prior to exceeding the 30-minute core uncovery time limit, then escalation to GE would not occur.

General Emergency - CG2.1

PNPS Basis:

When RPV level drops below -125 in., core uncovery is about to occur (ref. 1).

Four conditions are associated with a challenge to Primary Containment (PC) integrity:

- Containment closure is the action taken to secure either Primary Containment or Secondary Containment and the associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions (ref. 2). As applied to PNPS, containment closure is established when either Primary Containment integrity or Secondary Containment integrity is established in accordance with Section 3.7 of Technical Specifications (ref. 3).

- Deflagration (explosive) mixtures in the Primary Containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/SAMGs (Severe Accident Management Guides) indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to Primary Containment integrity. Hydrogen concentration of approximately 6% is considered the global deflagration concentration limit (ref. 4).

  Except for brief periods during plant startup and shutdown, oxygen concentration in the Primary Containment is maintained at insignificant levels by nitrogen inertion. The specified values for this threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen, ref. 3) and readily recognizable because 6% hydrogen is well above the EOP-3, *"Primary Containment Control"*, entry condition (ref. 5).

  The H2/O2 system monitors Primary Containment hydrogen (H2) and oxygen (O2) gas concentrations via the following containment penetration sample points: X-29E, X-106A-b, X-228J, X-15E, X-50A-d, and X-228C. The H2/O2 system returns its samples to Primary Containment via containment penetrations X-46F and X-228K.

The analyzer system consists of two separate units: an analyzer panel and a remote panel. Analyzer Panels C172 and C173 are located close to Primary Containment on the 74'3" elevation of the Reactor Building. Each panel contains all essential components necessary to perform the required containment atmosphere analysis, indication, and alarm functions. Recorders AR-1001-612A and AR-1001-612B and indicating meters are located in the Control Room in Panels C174 and C175. The backup Comsip H2/O2 analyzers, C172 and C173, are set to alarm at either 4% O2 or 4% H2 and are normally left in the "STANDBY" mode (ref. 6, 7, 8).

- Any unplanned increase in PC pressure in the Cold Shutdown or Refueling mode indicates a potential loss of containment closure capability. Unplanned Primary Containment pressure increases indicate containment closure cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.

- The Secondary Containment area radiation Maximum Safe Operating Values are indicative of problems in the Secondary Containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-4, *"Secondary Containment Control"*, Table L (ref. 9).

PNPS Basis Reference(s):

1. EOP-1, "*RPV Control*"
2. NEI 99-01 Definitions and Appendix C
3. Technical Specifications Section 3.7
4. BWROG EPG/SAG Revision 2, Sections PC/G
5. EOP-3, "*Primary Containment Control*"
6. PNPS 2.2.120, *"Postaccident Monitoring Panel"*
7. PNPS 2.2.133, *"H2/O2 Analyzer and C19 Systems"*
8. SAG-02, "*Containment and Radioactivity Release Control*"
9. EOP-4, "*Secondary Containment Control*"

General Emergency - CG2.1

EOP-4 Table L Secondary Containment Control Area Radiation Maximum Safe Operating Values

RADIATION LEVEL	VALUE (mR/hr)
NW equipment space/HPCI pump room - 17 ft 6 in. El.	1000
CRD pump room - 17 ft 6 in. El.	1000
RCIC pump room - 17 ft 6 in. El.	1000
SE equipment space - 17 ft 6 in. El.	1000
CRD HCU west area - 23 ft El.	1000
CRD HCU east area - 23 ft El.	1000
RB west area - 51 ft El.	1000
RB east area - 51 ft El.	1000
North Storage and laydown area - 74 ft 3 in. El. (H202)	1000
Fuel pool cooling pump/HX area - 74 ft 3 in. El	1000
SLC pump area - 91 ft 3 in. El.	1000
Skimmer surge tank area - 91 ft 3 in. El.	1000

General Emergency - CG2.2

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 2 - RPV Level

Initiating Condition: Loss of RPV inventory affecting fuel clad integrity with containment challenged

EAL:

CG2.2 General Emergency

RPV level **cannot** be monitored for ≥ 30 min. (Note 3) with a loss of inventory as indicated by **EITHER**:

Unexplained RPV leakage indication, Table C-1

OR

Erratic source range monitor indication

AND

Any containment challenge indication, Table C-5

Note 3: The Emergency Director should **not** wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

General Emergency - CG2.2

Table C-1 RPV Leakage Indications
- Drywell equipment drain sump level rise - Drywell floor drain sump level rise - Reactor Building equipment drain sump level rise - Reactor Building floor drain sump level rise - Torus level rise - RPV make-up rate rise - Observation of unisolable RCS leakage

Table C-5 Containment Challenge Indications
- Containment Closure **not** established (Note 4) - Deflagration concentrations exist inside PC - Unplanned rise in PC pressure - Secondary Containment area radiation > **any** Maximum Safe Operating Value (EOP-4, Table L)

Note 4: Containment closure is the action taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment closure is established when Primary or Secondary Containment integrity is established in accordance with Section 3.7 of Technical Specifications.

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

General Emergency - CG2.2

NEI 99-01 Basis:

This EAL represents the inability to restore and maintain RPV level to above the top of active fuel with containment challenged. Fuel damage is probable if RPV level cannot be restored as available decay heat will cause boiling, further reducing the RPV level. With the containment breached or challenged, 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. The GE is declared on the occurrence of the loss or imminent loss of function of all three barriers.

A number of variables can have a significant impact on heat removal capability challenging the fuel clad barrier. Examples include initial vessel level and shutdown heat removal system design.

Analysis indicates that core damage may occur within an hour following continued core uncovery; therefore, 30 minutes was conservatively chosen.

If containment closure is re-established prior to exceeding the 30-minute core uncovery time limit, then escalation to GE would not occur.

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.

PNPS Basis:

All Control Room RPV water level monitors have their indicated range referenced to an elevation of 482.5 inches above the bottom of the reactor vessel, 126.3 inches above the top of the active fuel (TAF). The common zero reference for RPV water level instrumentation was chosen as the bottom of the steam separators, 482.5 inches above the bottom of the reactor vessel. The low level instruments cover the level from 142" above vessel zero to 542" above vessel zero. The wide range instrument covers the upper 400" of the vessel from 425" above vessel zero to top of the vessel. Level Indicators LI-263-106A & B, LI-263-73A & B, LI-1001-650A & B, and Level Recorders LR-1001-604A & B (Pen 3) are used to monitor RPV level during accident conditions. Level Recorder LR-640-28 is used to monitor RPV level during shutdown conditions (ref. 2).

In preparation for refueling operations, a temporary shutdown/floodup transmitter is installed on Reactor Building 51' Rack C2206 and the narrow range feedwater level indicator LI-640-29B is replaced with a reactor shutdown/floodup digital level display with visual alarm points (EPIC point RXX054). This allows monitoring RPV level between "normal range" (instrument 0 to 466") through flood elevation (117') (ref. 1, 2).

Figure C-3 illustrates the elevations of the temporary RPV level floodup instrument (ref. 1).

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV. A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus level could be indicative of RHR valve misalignment or leakage. If the makeup rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory (ref. 3, 4, 5, 6, 7).

Four conditions are associated with a challenge to Primary Containment (PC) integrity:

- Containment closure is the action taken to secure either Primary Containment or Secondary Containment and the associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions (ref. 8). As applied to PNPS, containment closure is established when either Primary Containment integrity or Secondary Containment integrity is established in accordance with Section 3.7 of Technical Specifications (ref. 9).

- Deflagration (explosive) mixtures in the Primary Containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/SAMGs indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to Primary Containment integrity. Hydrogen concentration of approximately 6% is considered the global deflagration concentration limit (ref. 10).

  Except for brief periods during plant startup and shutdown, oxygen concentration in the Primary Containment is maintained at insignificant levels by nitrogen inertion. The specified values for this threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen, ref. 10) and readily recognizable because 6% hydrogen is well above the EOP-3, *"Primary Containment Control"*, entry condition (ref. 11).

  The H2/O2 system monitors Primary Containment hydrogen (H2) and oxygen (O2) gas concentrations via the following containment penetration sample points: X-29E, X-106A-b, X-228J, X-15E, X-50A-d, and X-228C. The H2/O2 system returns its samples to Primary Containment via containment penetrations X-46F and X-228K.

  The analyzer system consists of two separate units: an analyzer panel and a remote panel. Analyzer Panels C172 and C173 are located close to Primary Containment on the 74'3" elevation of the Reactor Building. Each panel contains all essential components necessary to perform the required containment atmosphere analysis, indication, and alarm functions. Recorders AR-1001-612A and AR-1001-612B and indicating meters are located in the Control Room in Panels C174 and C175. The backup Comsip H2/O2 analyzers, C172 and C173, are set to alarm at either 4% O2 or 4% H2 and are normally left in the "STANDBY" mode (ref. 12, 13, 14).

- Any unplanned increase in PC pressure in the Cold Shutdown or Refueling mode indicates a potential loss of containment closure capability. Unplanned Primary Containment pressure increases indicate containment closure cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.

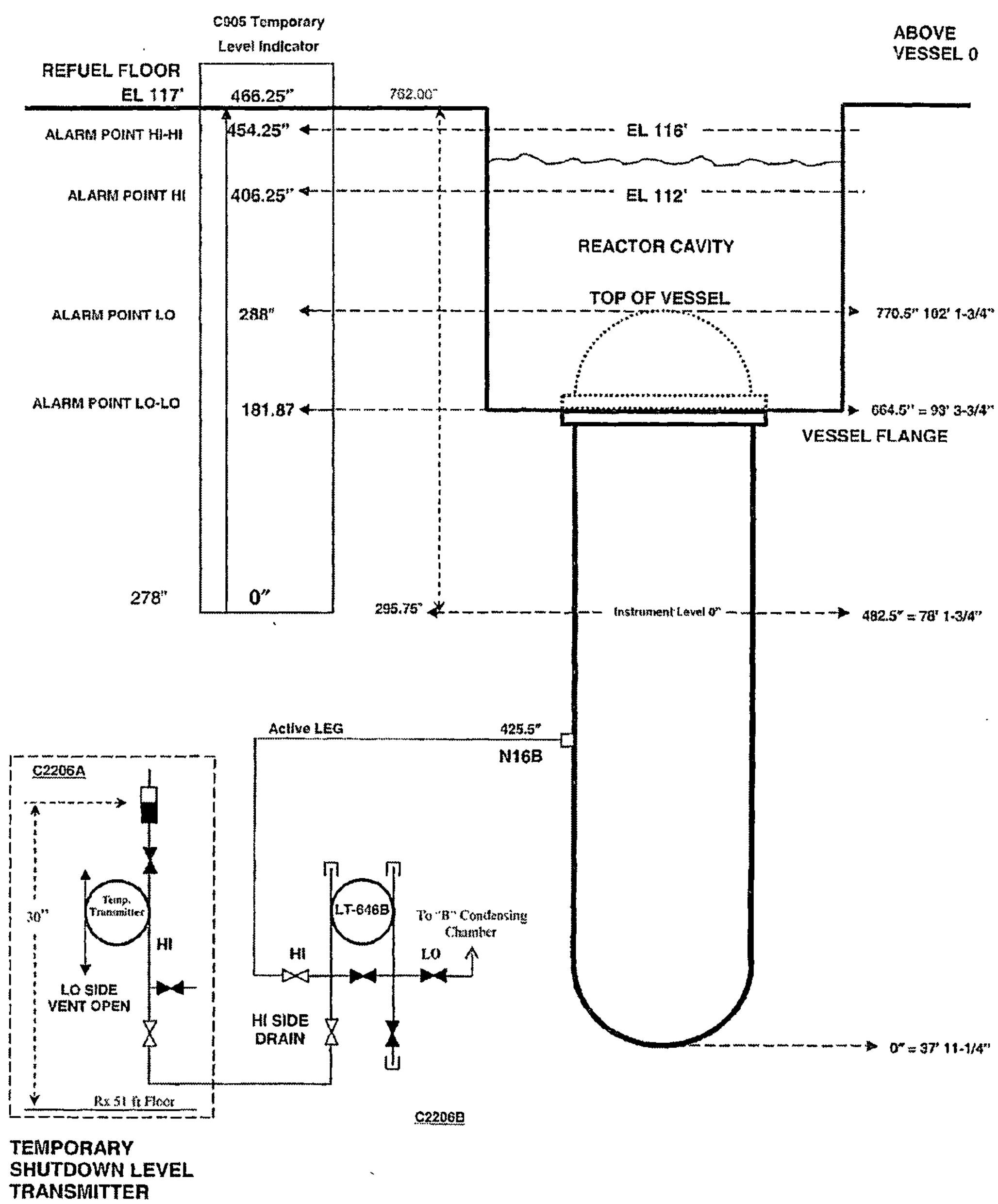
- The Secondary Containment area radiation Maximum Safe Operating Values are indicative of problems in the Secondary Containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-4, Secondary Containment Control, Table L (ref. 15).

PNPS Basis Reference(s):

1. PNPS 3.M.2-40, *"Refuel Outage Temporary Modification Reactor Shutdown/Flood-Up Level Indication"*
2. PNPS 2.2.80, *"Reactor Vessel Level, Temperature, and Internal Pressure Instrumentation"*
3. PNPS 2.2.77, *"Drywell Leak Detection Systems"*
4. PNPS 2.5.2.71, *"Radwaste Collection System"*
5. FSAR Section 4.10 - Nuclear System Leakage Rate Limits
6. FSAR Section 5.2 - Primary Containment System
7. FSAR Section 9.2 - Liquid Radwaste System
8. NEI 99-01 Definitions and Appendix C
9. Technical Specifications Section 3.7
10. BWROG EPG/SAG Revision 2, Sections PC/G
11. EOP-3, "*Primary Containment Control*"
12. PNPS 2.2.120, *"Postaccident Monitoring Panel"*
13. PNPS 2.2.133, *"H2/O2 Analyzer and C19 Systems"*
14. SAG-02, "*Containment and Radioactivity Release Control*"
15. EOP-4, "*Secondary Containment Control and Radioactivity Release Control*"

General Emergency - CG2.2

Figure C-3 Floodup RPV Level Indication



General Emergency - CG2.2

EOP-4 Table L Secondary Containment Control Area Radiation Maximum Safe Operating Values

RADIATION LEVEL	VALUE (mR/hr)
NW equipment space/HPCI pump room - 17 ft 6 in. El.	1000
CRD pump room - 17 ft 6 in. El.	1000
RCIC pump room - 17 ft 6 in. El.	1000
SE equipment space - 17 ft 6 in. El.	1000
CRD HCU west area - 23 ft El.	1000
CRD HCU east area - 23 ft El.	1000
RB west area - 51 ft El.	1000
RB east area - 51 ft El.	1000
North Storage and laydown area - 74 ft 3 in. El. (H202)	1000
Fuel pool cooling pump/HX area - 74 ft 3 in. El	1000
SLC pump area - 91 ft 3 in. El.	1000
Skimmer surge tank area - 91 ft 3 in. El.	1000

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 3 - RCS Temperature

Initiating Condition: Unplanned loss of decay heat removal capability with irradiated fuel in the RPV

EAL:

CU3.1 Unusual Event

Any unplanned event results in RCS temperature > 212°F due to loss of decay heat removal capability (Note 5)

Note 5: ILRT, CRD scram time testing, and hydrostatic testing in which RCS temperature is intentionally raised above 212°F in accordance with Technical Specifications LCO 3.14 are not applicable to these EALs.

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

NEI 99-01 Basis:

This EAL is 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 RPV inventory. Since the RCS usually remains intact in the cold shutdown mode, a large inventory of water is available to keep the core covered.

During refueling, the level in the RPV will normally be maintained above the RPV flange. Refueling evolutions that decrease water level below the RPV flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RCS temperatures depending on the time since shutdown.

Escalation to Alert would be via EAL CA2.1 based on an inventory loss or EAL CA3.1 based on exceeding its temperature criteria.

PNPS Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specifications cold shutdown temperature limit (212°F, ref. 1). These include (ref. 2):

- Reactor vessel shell adjacent to reactor vessel flange (TR-263-105 red pen on Panel C904 or TR-263-104 Pt. 2 Panel C921)
- Reactor vessel bottom head temperature TR-263-104 Pt. 9 Panel C921
- Reactor Recirculation Pump Loop A TR-260-151A on Panel C904
- Reactor Recirculation Pump Loop B TR-260-151B on Panel C904
- EPIC points RXX030, REC068, REC072, REC074, REC070, REC076, REC078

The Shutdown Cooling (SDC) mode of RHR is the primary means by which the reactor and internals are cooled and maintained below 212°F (depressurized) under shutdown conditions. The only other viable mechanisms for maintaining reactor coolant temperature < 212°F are to maximize RWCU cooling via the NRHX or to establish a feed/bleed with feed/condensate and RWCU reject. Should SDC be unavailable, reactor decay levels could potentially be well in excess of either of these other mechanisms. The inability to establish and maintain cold shutdown conditions under those conditions for which it is required poses a significant threat to reactor plant safety.

The note is a reminder that for certain planned and controlled events, RCS temperature may be raised above the LCO without changing the operating mode to hot conditions. For such events, the RCS temperature specified in the Technical Specifications definition of "cold shutdown" may be considered "NA" and operation considered not to be > 212°F or in hot shutdown mode while RCS temperature is above 212°F. With increased reactor vessel fluence over time, the minimum allowable vessel temperature increases at a given pressure. Periodic updates to the RPV P/T limit curves are performed as necessary based upon the results of analyses of irradiated surveillance specimens removed from the vessel. In the future it is expected that hydrostatic and leak testing may eventually be required with minimum reactor coolant temperatures exceeding 212°F. However, even with required minimum temperature requirements below 212°F, maintaining RCS temperature within a small band during the test can be impractical. Removal of the heat addition from recirculation pump operation and reactor core decay heat is coarsely controlled by control rod drive hydraulic system flow and reactor water cleanup system nonregenerative heat exchanger operation. Test conditions are focused on maintaining a steady state pressure, and tightly limited temperature control poses an unnecessary burden on the operator and may not be achievable in certain instances (ref. 4).

PNPS Basis Reference(s):

1. Technical Specifications, Section 1.0, Definitions
2. PNPS 2.1.7, *"Vessel Heatup and Cooldown"*
3. PNPS 2.4.25, *"Loss of Shutdown Cooling"*
4. Technical Specifications Section 3.14

Unusual Event - CU3.2

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 3 - RCS Temperature

Initiating Condition: Unplanned loss of decay heat removal capability with irradiated fuel in the RPV

EAL:

CU3.2	**Unusual Event**
Loss of **all** RCS temperature and RPV level indication for ≥ 15 min. (Note 3)	

Note 3: The Emergency Director should not wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

NEI 99-01 Basis:

This EAL is 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 RPV inventory. Since the RCS usually remains intact in the cold shutdown mode, a large inventory of water is available to keep the core covered.

During refueling the level in the RPV will normally be maintained above the RPV flange. Refueling evolutions that decrease water level below the RPV flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RPV temperatures depending on the time since shutdown.

Redundant means of RPV level indication are procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indications were to be lost in either the cold shutdown or refueling modes, this EAL would result in declaration of a UE if both temperature and level indications cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert would be via EAL CA2.1 based on an inventory loss or EAL CA3.1 based on exceeding its temperature criteria.

PNPS Basis:

All Control Room RPV water level monitors have their indicated range referenced to an elevation of 482.5 inches above the bottom of the reactor vessel, 126.3 inches above the top of the active fuel (TAF). The common zero reference for RPV water level instrumentation was chosen as the bottom of the steam separators, 482.5 inches above the bottom of the reactor vessel. The low level instruments cover the level from 142" above vessel zero to 542" above vessel zero. The wide range instrument covers the upper 400" of the vessel from 425" above vessel zero to top of the vessel. Level Indicators LI-263-106A & B, LI-263-73A & B, LI-1001-650A & B, and Level Recorders LR-1001-604A & B (Pen 3) are used to monitor RPV level during accident conditions. Level Recorder LR-640-28 is used to monitor RPV level during shutdown conditions (ref. 2).

In preparation for refueling operations, a temporary shutdown/floodup transmitter is installed on Reactor Building 51' Rack C2206 and the narrow range feedwater level indicator LI-640-29B is replaced with a reactor shutdown/floodup digital level display with visual alarm points (EPIC point RXX054). This allows monitoring RPV level between "normal range" (instrument 0 to 466") through flood elevation (117') (ref. 1, 2).

Figure C-3 illustrates the elevations of the temporary RPV level floodup instrument (ref. 1).

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specifications cold shutdown temperature limit (212°F, ref. 3). These include (ref. 4):

- Reactor vessel shell adjacent to reactor vessel flange (TR-263-105 red pen on Panel C904 or TR-263-104 Pt. 2 Panel C921)
- Reactor vessel bottom head temperature TR-263-104 Pt. 9 Panel C921
- Reactor Recirculation Pump Loop A TR-260-151A on Panel C904
- Reactor Recirculation Pump Loop B TR-260-151B on Panel C904
- EPIC points RXX030, REC068, REC072, REC074, REC070, REC076, REC078

PNPS Basis Reference(s):

1. PNPS 3.M.2-40, *"Refuel Outage Temporary Modification Reactor Shutdown/Flood-Up Level Indication"*
2. PNPS 2.2.80, *"Reactor Vessel Level, Temperature, and Internal Pressure Instrumentation"*
3. Technical Specifications, Section 1.0, Definitions
4. PNPS 2.1.7, *"Vessel Heatup and Cooldown"*

ATTACHMENT 9.5 **CATEGORY C, COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION**

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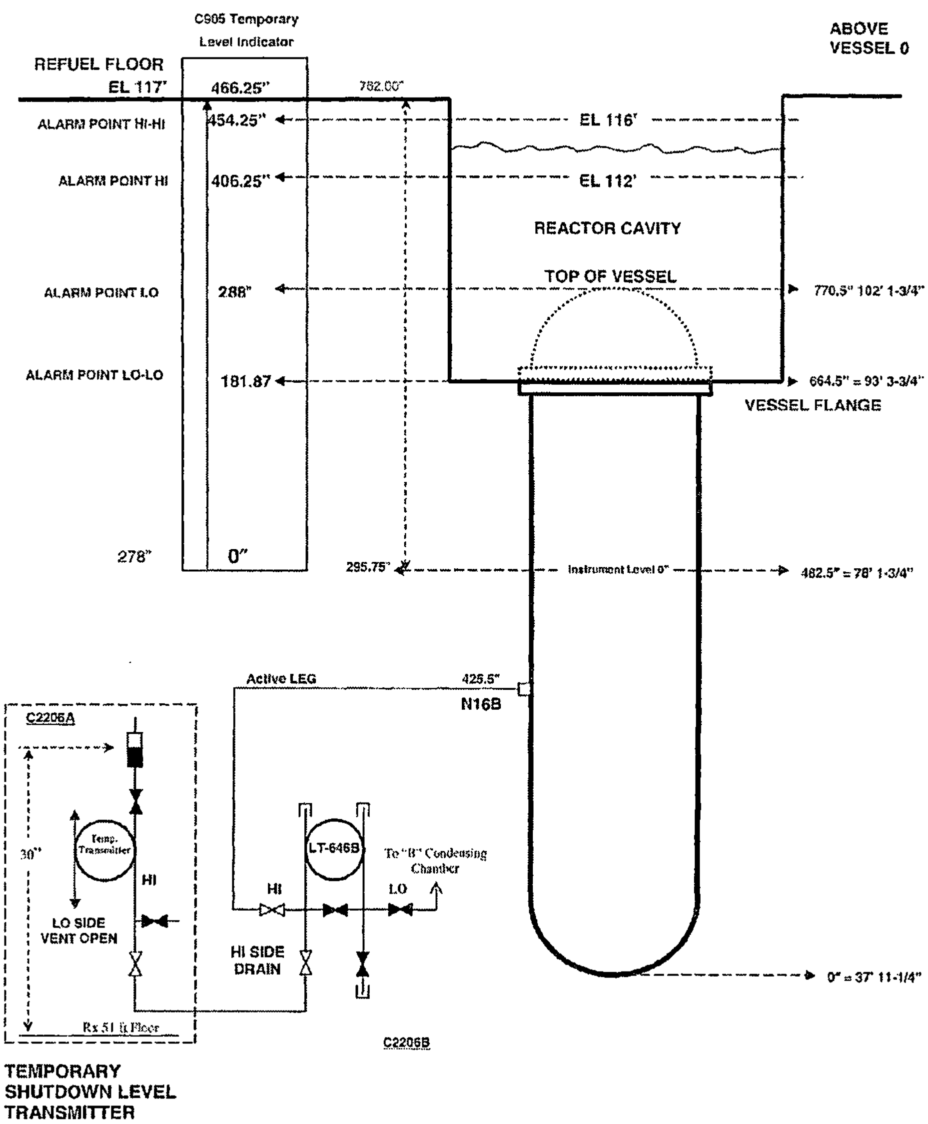
Unusual Event - CU3.2

FIGURE C-3 FLOODUP RPV LEVEL INDICATION



Alert - CA3.1

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 3 - RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.1 Alert

Any unplanned event results in RCS temperature > 212°F for > Table C-3 duration (Note 5)

OR

RPV pressure increase > 15 psig due to a loss of RCS cooling

Note 5: ILRT, CRD scram time testing and hydrostatic testing in which RCS temperature is intentionally raised above 212ºF in accordance with Technical Specifications LCO 3.14 are not applicable to these EALs

Table C-3 RCS Reheat Duration Thresholds	
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable	
1. RCS intact (Containment Closure N/A)	60 min.*
2. Containment Closure established **AND** RCS **not** intact	20 min.*
3. Containment Closure **not** established **AND** RCS **not** intact	0 min.

Alert - CA3.1

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

NEI 99-01 Basis:

The RCS Reheat Duration Thresholds table (Table C-3) addresses complete loss of functions required for core cooling for greater than 60 minutes during refueling and cold shutdown modes when RCS integrity is established. The 60-minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety.

The RCS Reheat Duration Thresholds table also addresses the complete loss of functions required for core cooling for greater than 20 minutes during refueling and cold shutdown modes when containment closure is established but RCS integrity is not established or RCS inventory is reduced. The allowed 20-minute time frame was included to allow operator action to restore the heat removal function if possible.

Finally, complete loss of functions required for core cooling during refueling and cold shutdown modes when neither containment closure nor RCS integrity are established. No delay time is allowed because the evaporated reactor coolant that may be released into the containment during this heatup condition could also be directly released to the environment.

The note (*) in the table indicates that this EAL is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the specified time frame.

The 15 psi pressure increase addresses situations where, due to high decay heat loads, the time provided to restore temperature control should be less than 60 minutes. The RCS pressure setpoint chosen is the lowest pressure that the site can read on installed control board instrumentation.

Escalation to Site Area Emergency would be via EAL CS2.1, CS2.2, or CS2.3 should boiling result in significant RPV level loss leading to core uncovery.

A loss of Technical Specifications components alone is not intended to constitute an Alert. The same is true of a momentary unplanned excursion above the Technical Specifications cold shutdown temperature limit 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 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.

PNPS Basis:

A 15 psig RPV pressure increase can be read on pressure indicators PI-640-25A/B and pressure recorder 6-PR-640-27 at Panel C905 (ref. 1).

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specifications cold shutdown temperature limit (212°F, ref. 2). These include (ref. 3):

- Reactor vessel shell adjacent to reactor vessel flange (TR-263-105 red pen on Panel C904 or TR-263-104 Pt. 2 Panel C921)
- Reactor vessel bottom head temperature TR-263-104 Pt. 9 Panel C921
- Reactor Recirculation Pump Loop A TR-260-151A on Panel C904
- Reactor Recirculation Pump Loop B TR-260-151B on Panel C904
- EPIC points RXX030, REC068, REC072, REC074, REC070, REC076, REC078

Containment closure is the action taken to secure either Primary Containment or Secondary Containment and the associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions (ref. 4). As applied to PNPS, containment closure is established when either Primary Containment integrity or Secondary Containment integrity is established in accordance with Section 3.7 of Technical Specifications (ref. 5)

Alert - CA3.1

The note is a reminder that any temperature increase above 212°F is an operating mode change from cold to hot conditions. Since each EAL is assigned one or more operating modes, the set of EALs that must be monitored must now include EALs associated with hot condition operating modes. The note is a reminder that for certain planned and controlled events RCS temperature may be raised above the LCO without changing the operating mode to hot conditions. For such events, the RCS temperature specified in the Technical Specifications definition of "Cold Shutdown" may be considered "NA" and operation considered not to be > 212°F or in Hot Shutdown mode while RCS temperature is above 212°F. With increased reactor vessel fluence over time, the minimum allowable vessel temperature increases at a given pressure. Periodic updates to the RPV P/T limit curves are performed as necessary based upon the results of analyses of irradiated surveillance specimens removed from the vessel. In the future it is expected that hydrostatic and leak testing may eventually be required with minimum reactor coolant temperatures exceeding 212°F. However, even with required minimum temperature requirements below 212°F, maintaining RCS temperature within a small band during the test can be impractical. Removal of the heat addition from recirculation pump operation and reactor core decay heat is coarsely controlled by control rod drive hydraulic system flow and reactor water cleanup system nonregenerative heat exchanger operation. Test conditions are focused on maintaining a steady state pressure, and tightly limited temperature control poses an unnecessary burden on the operator and may not be achievable in certain instances (ref. 6).

PNPS Basis Reference(s):

1. PNPS 8.M.2-6.1, *"Reactor Pressure Readout"*
2. Technical Specifications, Section 1.0, Definitions
3. PNPS 2.1.7, *"Vessel Heatup and Cooldown"*
4. NEI 99-01 Definitions and Appendix C
5. Technical Specifications Section 3.7
6. Technical Specifications Section 3.14

Unusual Event - CU4.1

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 4 - Communications

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

EAL:

CU4.1 Unusual Event

Loss of **all** Table C-2 onsite (internal) communication methods affecting the ability to perform routine operations

OR

Loss of **all** Table C-2 offsite (external) communication methods affecting the ability to perform offsite notifications

Table C-2 Communications Systems

System	Onsite (internal)	Offsite (external)
Plant Telephone System (CENTREX)	X	X
Wireless Telephone System	X	X
Pilgrim Station Radio System	X	
Plant Gaitronics System	X	
Alternate Shutdown Communication	X	
NRC-ENS Telephone, Direct Line		X
Satellite phones		X

Unusual Event - CU4.1

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel, Defueled

NEI 99-01 Basis:

The purpose of this IC and its associated EAL is to recognize a loss of communications capability that either defeats the plant operations staff's ability to perform routine tasks necessary for plant operations or the ability to communicate issues with offsite authorities. The loss of offsite communications ability is expected to be significantly more comprehensive than the condition addressed by 10CFR50.72.

The availability of one method of ordinary offsite communications is sufficient to inform federal, state, and local authorities of plant issues. 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.

PNPS Basis:

Onsite/offsite communications include one or more of the systems listed in Table C-2 (ref. 1, 2, 3, 4, 5). A description of the capabilities of each system is given in Section 4.0 of PNPS 2.2.17, *"Communications Systems"* (ref. 2).

This EAL is the cold condition equivalent of the hot condition EAL SU4.2.

PNPS Basis Reference(s):

1. FSAR Section 10.15
2. PNPS 2.2.17, *"Communications Systems"*
3. PNPS 2.4.57, *"Loss of Public-Address System"*
4. PNPS 8.A.13, *"Plant Emergency Alarms And Radio Test"*
5. EP-AD-413, *"Emergency Communications Test"*

Unusual Event - CU5.1

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 5 - Inadvertent Criticality

Initiating Condition: Inadvertent criticality

EAL:

CU5.1 Unusual Event

Unplanned sustained positive period observed on nuclear instrumentation

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

NEI 99-01 Basis:

This EAL addresses criticality events that occur in cold shutdown or refueling modes such as fuel misloading events. This EAL indicates a potential degradation of the level of safety of the plant, warranting a UE classification.

Escalation would be by Emergency Director judgment.

PNPS Basis:

Period meters NI-750-4A, C, B, and D on Panel C905 identify this condition as well as annunciator **"SRM PERIOD"** (C905L-G9). Amber lights on Panel C905 illuminate when its SRM channel period is less than 20 seconds (seal in) (ref. 1, 2).

PNPS Basis Reference(s):

1. PNPS 2.2.64, *"Source Range Monitoring System"*
2. PNPS 3.M.2-5.1, *"Source Range Monitor Calibration Instruction"*

Unusual Event - CU6.1

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 6 - Loss of DC Power

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

EAL:

CU6.1 Unusual Event

< 105V DC bus voltage indications on **all** Technical Specifications required 125V DC buses for ≥ 15 min. (Note 3)

Note 3: The Emergency Director should **not** wait until the applicable time has elapsed but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

NEI 99-01 Basis:

The purpose of this IC and its associated EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during cold shutdown or refueling operations.

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 in accordance with EAL CA3.1.

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

Unusual Event - CU6.1

PNPS Basis:

The essential 125V DC power distribution is illustrated in Figure C-2.

The loss of required essential 125V DC power poses a significant threat to decay heat removal capability and reactor plant safety. Operability of safety-related equipment and safety system protective functions are severely degraded. The 125V DC system provides power to CSCS initiation logics, controllers, and indications. The 125V DC system also provides control power and tripping power for high voltage AC protective devices. If not restored within a short period of time, significant system and equipment failures may be imminent depending upon plant conditions at the time of the loss.

Annunciators **"A 125V DC UNDERVOLTAGE"** (C3RC-A7) and **"B 125V DC UNDERVOLTAGE"** (C3RC-B7) alarm at 124V DC (decreasing) and signal loss of Panel D16 and D17, respectively.

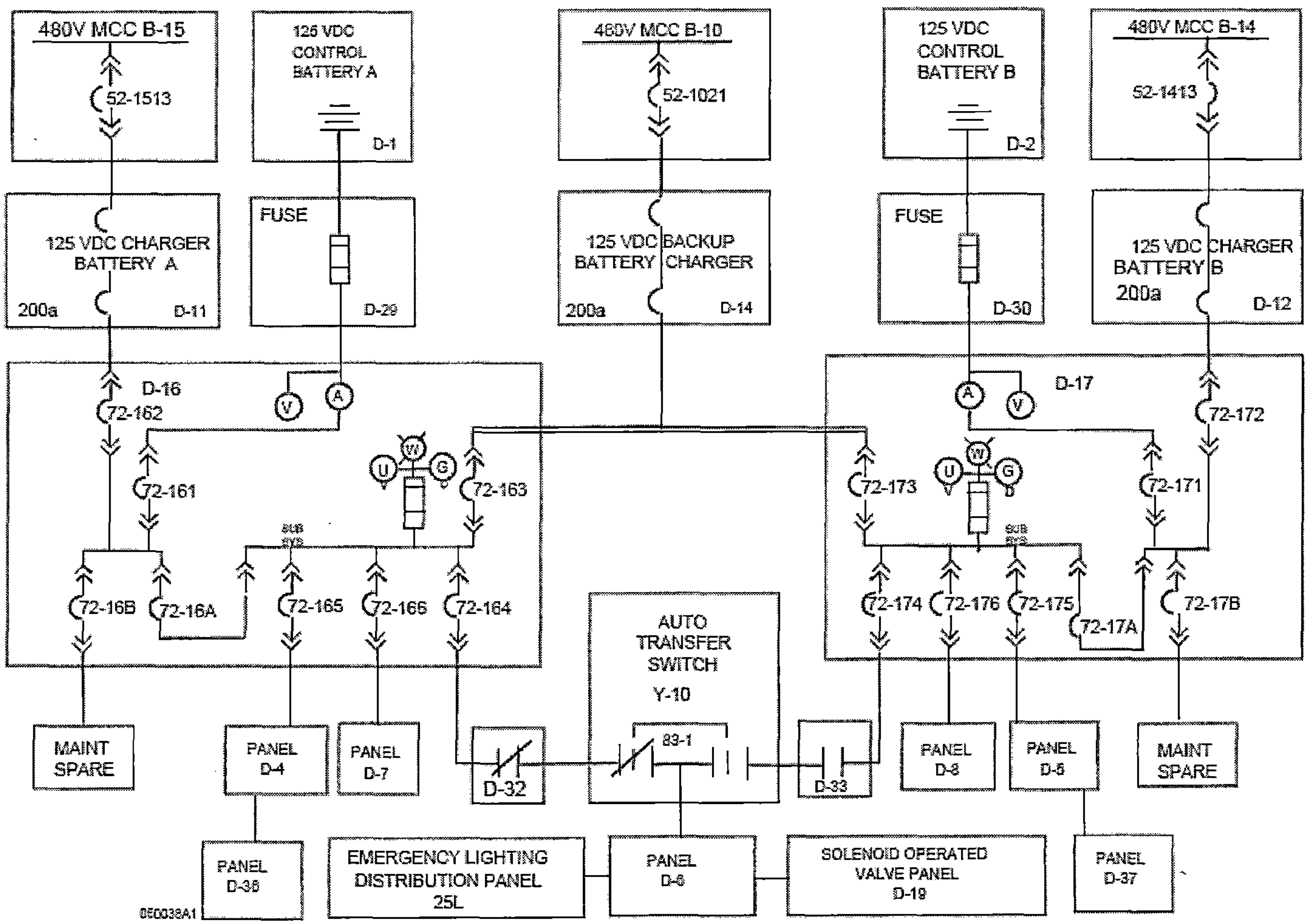
This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS7.1.

PNPS Basis Reference(s):

1. FSAR Figure 8.6-1
2. PNPS 2.2.14, *"125V DC Battery Systems"*
3. PNPS 5.3.11, *"Loss of Essential DC Bus D16 or D4 and D36"*
4. PNPS 5.3.12, *"Loss of Essential DC Bus D17 or D5 and D37"*
5. PNPS ARP C3RC-A7
6. PNPS ARP C3RC-B7

Unusual Event - CU6.1

Figure C-2 Essential 125V DC Power Distribution



Introduction

Table F-1 lists the threshold conditions that define the loss and potential loss of the three fission product barriers (fuel clad, Reactor Coolant System, and Primary Containment). The table is structured so that each of the three barriers occupies adjacent columns. Each fission product barrier column is further divided into two columns; one for loss thresholds and one for potential loss thresholds.

The first column of the table (to the left of the Fuel Clad Barrier Loss column) lists the categories (types) of fission product barrier thresholds. The fission product barrier categories are:

A. RPV Level

B. PC Pressure/Temperature

C. Isolation

D. Rad

E. Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word "None" is entered in the cell.

Thresholds are assigned sequential numbers to facilitate referencing. If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a loss or potential loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier loss/potential loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier loss and potential loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL user first scans down the category column of Table F-1, locates the likely category, and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine whether a threshold has been exceeded. If a threshold has not been exceeded, the EAL user proceeds to the next likely category and continues review of the thresholds in the new category

If the EAL user determines that any threshold has been exceeded, by definition the barrier is lost or potentially lost - even if multiple thresholds in the same barrier column are exceeded, only that one barrier is lost or potentially lost. The EAL user must examine each of the three fission product barriers to determine whether other barrier thresholds in the category are lost or potentially lost. For example, if containment radiation is sufficiently high, a loss of the fuel clad and RCS barriers and a potential loss of the Primary Containment barrier can occur. Barrier losses and potential losses are then applied to the algorithms given in EALs FG1.1, FS1.1, FA1.1, and FU1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the fuel clad barrier threshold bases appear first, followed by the RCS barrier, and finally the Primary Containment barrier threshold bases. In each barrier, the bases are given showing the "loss" threshold followed by the "potential loss" threshold beginning with Category A then sequentially to E.

ATTACHMENT 9.6 FISSION PRODUCT BARRIER LOSS/POTENTIAL LOSS MATRIX AND TECHNICAL BASES

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Table F-1 Fission Product Barrier Matrix

	Fuel Clad Barrier		Reactor Coolant System Barrier		Primary Containment Barrier	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A. RPV Level	1. Primary Containment Flooding required due to **EITHER:** RPV water level cannot be restored and maintained above -150 in. (MSCRWL) **OR** RPV water level cannot be determined and core damage is occurring	5. RPV level **cannot** be restored and maintained > -125 in. or **cannot** be determined	7. RPV level cannot be restored and maintained > -125 in. or **cannot** be determined	None	None	22. SAG entry required
B. PC Pressure / Temperature	None	None	8. PC pressure > 2.2 psig due to RCS leakage	None	16. PC pressure rise followed by a rapid unexplained drop in PC pressure 17. PC pressure response **not** consistent with LOCA conditions	23. Torus bottom pressure > 60 psig and rising 24. Deflagration concentrations exist inside PC 25. Torus water temperature and RPV pressure **cannot** be maintained below Heat Capacity Temperature Limit (EOP-11 Figure 2)
C. Isolation	None	None	9. Release pathway exists outside primary containment resulting from isolation failure in **any** of the following (excluding normal process system flowpaths from an unisolable system): - Main steam line - HPCI steam line - RCIC steam line - RWCU - Feedwater 10. Emergency RPV depressurization required	13. RCS leakage > 50 gpm inside the drywell 14. Unisolable primary system discharge outside primary containment **AND** A valid entry condition to EOP-4 exists due to Secondary Containment area radiation or temperature above **any** Maximum Normal Operating Value (EOP-4, Table H)	18. Failure of **any** valve in **any** one line to close **AND** Direct downstream pathway to the environment exists after PC isolation signal 19. Intentional PC venting per EOPs 20. Unisolable primary system discharge outside PC resulting in Secondary Containment area radiation or temperature above **any** Maximum Safe Operating Value (EOP-4, Table L)	None
D. Rad	2. Drywell High Range Area Radiation Monitor (RIT-1001-606A and B) > 800 R/hr OR Torus High Range Area Radiation Monitor (RIT-1001-607A and B) > 50 R/hr 3. Primary coolant activity > 300 µCi/gm I-131 dose equivalent	None	11. Drywell High Range Area Radiation Monitor (RIT-1001-606A and B) > 65 R/hr **OR** Torus High Range Area Radiation Monitor (RIT-1001-607A and B) > 4 R/hr	None	None	26. Drywell High Range Area Radiation Monitor (RIT-1001-606A and B) > 8,000 R/hr **OR** Torus High Range Area Radiation Monitor (RIT-1001-607A and B) > 500 R/hr
E. Judgment	4. **Any** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier	6. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier	12. **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	15. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	21. **Any** condition in the opinion of the Emergency Director that indicates loss of the PC barrier	27. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the PC barrier

Fuel Clad
RPV Level - Loss

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Loss

Threshold:

1. Primary Containment Flooding required due to **EITHER** of the following:
   - RPV water level cannot be restored and maintained above -150 in. (MSCRWL)
   - RPV water level cannot be determined and core damage is occurring

NEI 99-01 Basis:

This site-specific value corresponds to the level used in EOPs to indicate challenge of core cooling. This is the minimum value to assure core cooling without further degradation of the clad.

Fuel Clad
RPV Level - Loss

PNPS Basis:

Severe Accident Guidelines (SAGs) are entered from EOP-1, EOP-2, EOP-3, EOP-16, and EOP-26 when Primary Containment flooding is required. Primary Containment flooding is required when core cooling is severely challenged. These EOPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined. SAG entry is required when (ref. 1):

- RPV water level cannot be restored and maintained above -150 in. (MSCRWL) (ref. 1).
- RPV water level cannot be determined and core damage is occurring (ref. 2, 3).

The above EOP conditions represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

This threshold is also a potential loss of the Primary Containment barrier (PC P-Loss A.22). Since SAG entry occurs after core uncovery has occurred, a loss of the RCS barrier exists (RCS Loss A.7). SAG entry, therefore, represents a loss of two barriers and a potential loss of a third, which requires a General Emergency classification.

PNPS Basis Reference(s):

1. EOP-1, *"RPV Control"*
2. EOP-16, *"RPV Flooding"*
3. EOP-26, "*RPV Flooding, Failure-To-Scram"*
4. EOP-2, "*RPV Control, Failure-To-Scram*"
5. EOP-3, *"Primary Containment Control"*

Fuel Clad
RPV Level - Potential Loss

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

5. RPV level **cannot** be restored and maintained > -125 in. or **cannot** be determined

NEI 99-01 Basis:

This threshold is the same as the RCS barrier loss threshold and corresponds to the site-specific water level at the top of the active fuel. Thus, this threshold indicates a potential loss of the fuel clad barrier and a loss of RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

PNPS Basis:

An RPV level instrument reading of -125 in. indicates RPV level is at the top of active fuel (TAF) (ref. 1). When RPV level is at or above TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme RPV level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops to TAF, the level is indicative of a challenge to core cooling and the fuel clad barrier.

Fuel Clad
RPV Level - Potential Loss

When RPV level cannot be determined, EOPs require entry to EOP-16, *"RPV Flooding"*, or EOP-26, *"RPV Flooding, Failure-to-Scram"*. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained (ref. 2, 3). When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-16/26 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold the Minimum Steam Cooling Pressures (in scram-failure events) (ref. 2, 3). If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the fuel clad barrier exists.

Note that EOP-2, *"RPV Control, Failure-to-Scram"*, may require intentionally lowering RPV water level to TAF and controlling level between the Minimum Steam Cooling RPV Water Level (MSCRWL) and TAF (ref. 4). Under these conditions, a high-power ATWS event exists and requires at least a Site Area Emergency classification in accordance with the System Malfunction - ATWS/Criticality EALs.

If core uncovery is threatened, the EOPs specify alternate, more extreme RPV level control measures in order to restore and maintain adequate core cooling, including depressurization and restoration with low level pumps.

Determination of "restore and maintain" is based on the actions driven by Emergency Operating Procedures to restore level. The inability to reverse the RPV level lowering trend after lining up injection sources and injecting, including the use of low pressure systems following an Emergency Depressurization, would warrant classification.

PNPS Basis Reference(s):

1. EOP-1, *"RPV Control"*
2. EOP-16, *"RPV Flooding"*
3. EOP-26, *"RPV Flooding, Failure-To-Scram"*
4. EOP-2, *"RPV Control, Failure-To-Scram"*

Fuel Clad
PC Pressure/Temp - Loss

Barrier: Fuel Clad

Category: B. PC Pressure/Temperature

Degradation Threat: Loss

Threshold:

None

Fuel Clad
PC Pressure/Temp - Potential Loss

Barrier: Fuel Clad

Category: B. PC Pressure/Temperature

Degradation Threat: Potential Loss

Threshold:

None

Fuel Clad
Isolation - Loss

Barrier: Fuel Clad

Category: C. Isolation

Degradation Threat: Loss

Threshold:

None

Fuel Clad
Isolation - Potential Loss

Barrier: Fuel Clad

Category: C. Isolation

Degradation Threat: Potential Loss

Threshold:

None

Fuel Clad
Rad - Loss

Barrier: Fuel Clad

Category: D. Rad

Degradation Threat: Loss

Threshold:

2. Drywell High Range Area Radiation Monitor (RIT-1001-606A and B) > 800 R/hr

   OR

   Torus High Range Area Radiation Monitor (RIT-1001-607A and B) > 50 R/hr

NEI 99-01 Basis:

This threshold reading is a value which indicates the release of reactor coolant with elevated activity indicative of fuel damage, into the drywell.

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 threshold. Thus, this threshold indicates a loss of both fuel clad barrier and RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

There is no potential loss threshold associated with this item.

Fuel Clad
Rad - Loss

PNPS Basis:

The Drywell and Torus High Range Area Radiation Monitor readings indicate the release of reactor coolant into the Primary Containment with elevated activity indicative of fuel damage. The readings were derived assuming (ref. 1):

- The reactor has been shutdown for 1 hour
- 2% fuel clad damage
- No drywell sprays in operation
- A LOCA-depressurized system
- The instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory into the drywell atmosphere

The monitor reading of 1 hour after shutdown provides a realistic time for a degraded fuel condition to develop in a fast-breaking accident, which results in conservative threshold values for releases to Primary Containment with less than 1 hour after shutdown (ref. 1).

In order to reach this fuel clad barrier loss threshold, a loss of the RCS barrier has already occurred (see RCS Loss D.11). This threshold, therefore, represents at least a Site Area Emergency classification.

PNPS Basis Reference(s):

1. EP-IP-330, *"Core Damage"*

Fuel Clad
Rad - Loss

Barrier: Fuel Clad

Category: D. Rad

Degradation Threat: Loss

Threshold:

3. Primary coolant activity > 300 μCi/gm I-131 dose equivalent

NEI 99-01 Basis:

The reactor coolant activity of 300 μCi/gm I-131 equivalent is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the fuel clad barrier is considered lost.

There is no potential loss threshold associated with this item.

PNPS Basis:

None

PNPS Basis Reference(s):

1. NEI 99-01 Revision 5

Fuel Clad
Rad - Potential Loss

Barrier: Fuel Clad

Category: D. Rad

Degradation Threat: Potential Loss

Threshold:

None

Fuel Clad
Judgment - Loss

Barrier: Fuel Clad

Category: E. Judgment

Degradation Threat: Loss

Threshold:

4. **Any** condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier

NEI 99-01 Basis:

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the fuel clad barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

PNPS Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining whether the fuel clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability, and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within 2 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.

Fuel Clad
Judgment - Loss

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

The "Judgment" classifications should not be used if an applicable EAL classification has already been determined as included in the EAL chart.

PNPS Basis Reference(s):

None

Fuel Clad
Judgment - Potential Loss

Barrier: Fuel Clad

Category: E. Judgment

Degradation Threat: Potential Loss

Threshold:

6. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier

NEI 99-01 Basis:

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

PNPS Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining whether the fuel clad barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability, and dominant accident sequences.

- <u>Imminent barrier degradation</u> exists if the degradation will likely occur within 2 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.

- <u>Barrier monitoring</u> 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.

Fuel Clad
Judgment - Potential Loss

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

The "Judgment" classifications should not be used if an applicable EAL classification has already been determined as included in the EAL chart.

PNPS Basis Reference(s):

None

Reactor Coolant System
RPV Level - Loss

Barrier: Reactor Coolant System

Category: A. RPV Level

Degradation Threat: Loss

Threshold:

7. RPV level **cannot** be restored and maintained > -125 in. or **cannot** be determined

NEI 99-01 Basis:

The loss threshold for RPV water level corresponds to the level that is used in EOPs to indicate challenge of core cooling.

This threshold is the same as the fuel clad barrier potential loss threshold and corresponds to a challenge to core cooling. Thus, this threshold indicates a loss of RCS barrier and potential loss of fuel clad barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

There is no potential loss threshold associated with this item.

PNPS Basis:

An RPV level instrument reading of -125 in. indicates RPV level is at the top of active fuel (TAF) (ref. 1). TAF is significantly lower than the normal operating RPV level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Primary Containment barriers and initiation of all ECCS. If RPV level cannot be maintained above TAF, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the decreasing level trend. The cause of the loss of RPV inventory is therefore assumed to be a LOCA. By definition, a LOCA event is a loss of the RCS barrier.

Reactor Coolant System
RPV Level - Loss

When RPV level cannot be determined, EOPs require entry to EOP-16, *"RPV Flooding"*, or EOP-26, *"RPV Flooding, Failure-to-Scram"*. RPV water level indication provides the primary means of knowing whether adequate core cooling is being maintained (ref. 2, 3). When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-16/26 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold the Minimum Steam Cooling Pressures (in scram-failure events) (ref. 2, 3). If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the fuel clad barrier exists.

Note that EOP-2, *"RPV Control, Failure-to-Scram"*, may require intentionally lowering RPV water level to TAF and controlling level between the Minimum Steam Cooling RPV Water Level (MSCRWL) and TAF (ref. 4). Under these conditions, a high-power ATWS event exists and requires at least a Site Area Emergency classification in accordance with the System Malfunction - ATWS/Criticality EALs.

If core uncovery is threatened, the EOPs specify alternate, more extreme RPV level control measures in order to restore and maintain adequate core cooling, including depressurization and restoration with low level pumps.

Determination of "restore and maintain" is based on the actions driven by Emergency Operating Procedures to restore level. The inability to reverse the RPV level lowering trend after lining up injection sources and injecting, including the use of low pressure systems following an Emergency Depressurization, would warrant classification.

PNPS Basis Reference(s):

1. EOP-1, *"RPV Control"*
2. EOP-16, *"RPV Flooding"*
3. EOP-26, "*RPV Flooding, Failure-To-Scram*"
4. EOP-2, "*RPV Control, Failure-To-Scram*"

Reactor Coolant System
RPV Level - Potential Loss

Barrier: Reactor Coolant System

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

None

Reactor Coolant System
PC Pressure/Temp - Loss

Barrier: Reactor Coolant System

Category: B. PC Pressure/Temperature

Degradation Threat: Loss

Threshold:

8. PC pressure > 2.2 psig due to RCS leakage

NEI 99-01 Basis:

The loss threshold for Primary Containment pressure is based on the drywell high pressure setpoint which indicates a LOCA by automatically initiating the ECCS or equivalent makeup system.

There is no potential loss threshold associated with this item.

PNPS Basis:

The drywell high pressure scram setpoint is an entry condition to EOP-1, *"RPV Control"*, and EOP-3, *"Primary Containment Control"* (ref. 1, 2, 3). Normal Primary Containment pressure control functions (e.g., operation of drywell cooling, SGTS, etc.) are specified in EOP-3 in advance of less desirable but more effective functions (e.g., operation of drywell or torus sprays, etc.).

In the PNPS design basis, Primary Containment pressures above the drywell high pressure scram setpoint are assumed to be the result of a high-energy release into the containment for which normal pressure control systems are inadequate or incapable of reversing the increasing pressure trend. Pressures of this magnitude, however, can be caused by non-LOCA events such as a loss of drywell cooling or inability to control Primary Containment vent/purge (ref. 4).

Reactor Coolant System
PC Pressure/Temp - Loss

The threshold phrase "...due to RCS leakage" focuses the barrier failure on the RCS instead of the non-LOCA malfunctions that may adversely affect Primary Containment pressure. PC pressure greater than 2.2 psig with corollary indications (drywell temperature, humidity, etc.) should therefore be considered a loss of the RCS barrier. Loss of drywell cooling that results in pressure greater than 2.2 psig should not be considered an RCS barrier loss.

PNPS Basis Reference(s):

1. EOP-1 "*RPV Control,*" Entry Condition
2. EOP-3 "*Primary Containment Control,*" Entry Condition
3. PNPS Technical Specifications 3.1
4. PNPS 2.4.44, "*Loss of Drywell Area Coolers*"

Reactor Coolant System
PC Pressure/Temp - Potential Loss

Barrier: Reactor Coolant System

Category: B. PC Pressure/Temperature

Degradation Threat: Potential Loss

Threshold:

None

Reactor Coolant System Isolation - Loss

Barrier: Reactor Coolant System

Category: C. Isolation

Degradation Threat: Loss

Threshold:

9. Release pathway exists outside Primary Containment resulting from isolation failure in **any** of the following (excluding normal process system flow paths from an unisolable system):
   - Main steam line
   - HPCI steam line
   - RCIC steam line
   - RWCU
   - Feedwater

NEI 99-01 Basis:

An unisolable MSL break is a breach of the RCS barrier. Thus, this threshold is included for consistency with the Alert emergency classification level.

Other large high-energy line breaks such as HPCI, Feedwater, RWCU, or RCIC that are unisolable also represent a significant loss of the RCS barrier and should be considered as MSL breaks for purposes of classification.

Reactor Coolant System
Isolation - Loss

PNPS Basis:

The conditions of this threshold include required containment isolation failures allowing a flow path to the environment. A release pathway outside Primary Containment exists when flow is not prevented by downstream isolations. In the case of a failure of both isolation valves to close but in which no downstream flow path exists, emergency declaration under this threshold would not be required. Similarly, if the emergency response requires the normal process flow of a system outside Primary Containment (e.g., EOP requirement to bypass MSIV low RPV water level interlocks and maintain the main condenser as a heat sink using main turbine bypass valves), the threshold is not met. The combination of these threshold conditions represent the loss of both the RCS and Primary Containment (see PC Loss C.18) barriers and justifies declaration of a Site Area Emergency (i.e., loss or potential loss of any two barriers).

Even though RWCU and Feedwater systems do not contain steam, they are included in the list because an unisolable break could result in the high pressure discharge of fluid that is flashed to steam from relatively large volume systems directly connected to the RCS.

PNPS Basis Reference(s):

1. FSAR Sections 4.4, 4.5, 4.6, 4.11 (MSL)
2. FSAR Section 6.3 (HPCI)
3. FSAR Section 4.7 (RCIC)
4. FSAR Section 4.9 (RWCU)
5. FSAR Section 11.9 (FW)
6. PNPS 2.2.21, *"High Pressure Coolant Injection System (HPCI)"*
7. PNPS 2.2.22, *"Reactor Core Isolation Cooling System (RCIC)"*
8. PNPS 2.2.83, *"Reactor Cleanup System"*
9. PNPS 2.2.92, *"Main Steam Line Isolation and Turbine Bypass Valves"*

Reactor Coolant System Isolation - Loss

Barrier: Reactor Coolant System

Category: C. Isolation

Degradation Threat: Loss

Threshold:

10. Emergency RPV depressurization required

NEI 99-01 Basis:

Plant symptoms requiring emergency RPV depressurization in accordance with PNPS EOPs are indicative of a loss of the RCS barrier. If emergency RPV depressurization is required, the plant operators are directed to open safety relief valves (SRVs) and keep them open. Even though the RCS is being vented into the suppression pool, a loss of the RCS should be considered to exist due to the diminished effectiveness of the RCS pressure barrier to a release of fission products beyond its boundary.

PNPS Basis:

Plant symptoms requiring emergency RPV depressurization (RPV-ED) are specified in the EOPs (ref. 1, 2, 3, 4, 5, 6, 7, 8, 9, 10). If emergency RPV depressurization is required, the plant operators are directed to open SRVs and keep them open regardless of any subsequent radiological release rate (ref. 7, 8, 9, 10).

Reactor Coolant System Isolation - Loss

PNPS Basis Reference(s):

1. EOP-1, *"RPV Control"*
2. EOP-2, *"RPV Control, Failure-to-Scram"*
3. EOP-3, *"Primary Containment Control"*
4. EOP-4, *"Secondary Containment Control"*
5. EOP-5, *"Radioactivity Release Control"*
6. EOP-8, *"Steam Cooling"*
7 EOP-16, *"RPV Flooding"*
8. EOP-17, *"Emergency RPV Depressurization"*
9. EOP-26, *"RPV Flooding, Failure-to-Scram"*
10. EOP-27, *"Emergency RPV Depressurization, Failure-to-Scram"*

Reactor Coolant System Isolation - Potential Loss

Barrier: Reactor Coolant System

Category: C. Isolation

Degradation Threat: Potential Loss

Threshold:

13. RCS leakage > 50 GPM inside the drywell

NEI 99-01 Basis:

This threshold is based on leakage 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.

If primary system leak rate information is unavailable, other indicators of RCS leakage should be used.

PNPS Basis:

RCS leakage inside the drywell is normally determined by monitoring drywell equipment and floor drain sump pump-out rates. This method of monitoring leakage may be isolated as part of the drywell isolation and thus may be unavailable. If primary system leak rate information is unavailable, other indicators of RCS leakage should be used (ref. 1). Inventory loss events, such as a stuck open SRV, should not be considered when referring to "RCS leakage" because they are not indications of a break which could propagate.

PNPS Basis Reference(s):

1. PNPS 2.2.77, *"Drywell Leak Detection Systems"*

Reactor Coolant System
Isolation - Potential Loss

Barrier: Reactor Coolant System

Category: C. Isolation

Degradation Threat: Potential Loss

Threshold:

14. Unisolable primary system discharge outside Primary Containment

    AND

    A valid entry condition to EOP-4 exists due to Secondary Containment area radiation or temperature above **any** Maximum Normal Operating Value (EOP-4, Table H)

NEI 99-01 Basis:

Potential loss of RCS based on primary system leakage outside the Primary Containment is determined from site-specific temperature or area radiation Max Normal setpoints 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 reaching the threshold barriers and confirmed to be caused by RCS leakage warrant 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 loss threshold C.20 (after a containment isolation) and a General Emergency when the fuel clad barrier criteria are also exceeded.

Reactor Coolant System
Isolation - Potential Loss

PNPS Basis:

The presence of elevated general area temperatures or radiation levels in the Secondary Containment may be indicative of unisolable primary system leakage outside the Primary Containment. The Maximum Normal Operating Values (Table F-2 below) define this RCS threshold because they signify the onset of abnormal system operation. When parameters reach this level, equipment failure or misoperation may be occurring. Elevated parameters may also adversely affect the ability to gain access to or operate equipment within the affected area. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-4, Secondary Containment Control, Table H (ref. 1).

In general, multiple indications should be used to determine whether a primary system is discharging outside Primary Containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the Secondary Containment since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g., room flooding, high area temperatures, reports of steam in the Secondary Containment, an unexpected rise in feedwater flow rate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the Secondary Containment. For example, a high RWCU area temperature may be indicative of increased ambient temperatures due to seasonal changes that simply indicate repositioning of ventilation dampers is needed. Although a Table H temperature has reached its Maximum Normal Operating Value, the Shift Manager determines the entry condition to EOP-4 to be invalid and need not execute EOP-4. For such conditions, this EAL threshold is not met.

PNPS Basis Reference(s):

1. EOP-4, *"Secondary Containment Control"*

Reactor Coolant System Isolation - Potential Loss

Table F-2 Secondary Containment Area Temperature and Radiation Maximum Normal Operating Values

RADIATION LEVEL	VALUE (*mR/hr*)
NW equipment space/HPCI pump room - 17 ft 6 in. El. CRD pump room - 17 ft 6 in. El. RCIC pump room - 17 ft 6 in. El. SE equipment space - 17 ft 6 in. El. CRD HCU west area - 23 ft El. CRD HCU east area - 23 ft El. RB west area - 51 ft El. RB east area - 51 ft El. North Storage and laydown area - 74 ft 3 in. El. (H202 Fuel pool cooling pump/HX area - 74 ft 3 in. El SLC pump area - 91 ft 3 in. El. Skimmer surge tank area - 91 ft 3 in. El.	Unexplainable Loose Surface Contamination or Radiation, procedure 5.3.33

TEMPERATURE		VALUE (°F)
RWCU Holding Pump Area - 74 ft El.	TE-1291-60C	105
RWCU Filter Area - 74 ft El.	TE-1291-60A	105
RWCU Backwash Tank Area - 51 ft El.	TE-1291-60B	105
RWCU Pump "A" Room - 51 ft El.	TE-1291-60D	105
RWCU Pump "B" Room - 51 ft El.	TE-1291-60E	105
RWCU Heat Exchanger Room - 51 ft El. *	TE-1291-60F	115
RCIC Piping Area - Torus Compt *	TE-1360-23A	105
RCIC Turbine Area - Stairwell	TE-1360-23B	115
HPCI Piping Area - Torus Compt	TE-2374A	105
HPCI Turbine Area - 17 ft El.	TE-2374B	115
RWCU Piping Area - 36 ft El. Mezzanine	TE-1291-60H	105
RCIC Tip Room - 23 ft El.	TE-1360-23C	105
Main Steam Tunnel - 23 ft El.	TE-261-22A	140
HPCI Piping Area - 23 ft El. ("B" RHR Valve Room)	TE-2374C	105
RHR "B" & "D" Pump Area - Stairwell	TE-1001-92A	115
RHR "A" & "C" Pump Area - 6 ft El.	TE-1001-92B	115
RHR "A" & "C" Pump Area - Pipewell	TE-1001-92G	105
RWCU & RHR Valve Room - 23 ft El. ("A" RHR Vlv Rm)	TE-1001-92F	105
RHR Fuel Pool Heat Exchanger Room - 74 ft El. *	TE-1001-92H	105
HPCI Compartment H&V Cooler (Panel C-61B) *	24-TI-H-38/39	100
RHR A Quadrant (SE) H&V Cooler (Panel C-61A) *	24-TI-H-34/36	100
RHR B Quadrant (NW) H&V Cooler (Panel C-61A) *	24-TI-H-35/37	100
CRD Quadrant (NE) H&V Cooler (Panel C-61A)	24-TI-H-42/43	100
RCIC Quadrant (SW) H&V Cooler (Panel C-61A) *	24-TI-H-40/41	100
* Readings available on Kaye Computer		

Reactor Coolant System
Rad - Loss

Barrier: Reactor Coolant System

Category: D. Rad

Degradation Threat: Loss

Threshold:

11. Drywell High Range Area Radiation Monitor (RIT-1001-606A and B) > 65 R/hr

    OR

    Torus High Range Area Radiation Monitor (RIT-1001-607A and B) > 4 R/hr

NEI 99-01 Basis:

The loss threshold reading is a value which indicates the release of reactor coolant to the Primary Containment.

This reading will be less than that specified for fuel clad barrier loss threshold D.2. Thus, this threshold would be indicative of an RCS leak only. If the radiation monitor reading increased to that value specified by fuel clad barrier threshold, fuel damage would also be indicated.

There is no potential loss threshold associated with this item.

PNPS Basis:

The drywell threshold is based on the Technical Specifications maximum allowable coolant activity uniformly dispersed into the Primary Containment 1 hour after reactor shutdown. The corresponding torus radiation threshold is 0.5 R/hr. A value of 4 R/hr has been selected, however, to provide a readable on-scale indication. The Drywell and Torus High Range Area Radiation Monitor range is 1 to 1E7 R/hr.

Reactor Coolant System
Rad - Loss

PNPS Basis Reference(s):

1. EP-IP-330, *"Core Damage"*

Reactor Coolant System
Rad - Potential Loss

Barrier: Reactor Coolant System

Category: D. Rad

Degradation Threat: Potential Loss

Threshold:

None

Reactor Coolant System
Judgment - Loss

Barrier: Reactor Coolant System

Category: E. Judgment

Degradation Threat: Loss

Threshold:

12. **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier

NEI 99-01 Basis:

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost. In addition, the inability to monitor the barrier should also be considered in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

PNPS Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining whether the RCS barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability, and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within 2 hours based on a projection of current safety system performance. The term "imminent" refers to the 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.

Reactor Coolant System
Judgment - Loss

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

The "Judgment" classifications should not be used if an applicable EAL classification has already been determined as included in the EAL chart.

PNPS Basis Reference(s):

None

Reactor Coolant System
Judgment - Potential Loss

Barrier: Reactor Coolant System

Category: E. Judgment

Degradation Threat: Potential Loss

Threshold:

15. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier

NEI 99-01 Basis:

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is potentially lost. In addition, the inability to monitor the barrier should also be considered in this threshold as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

PNPS Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining whether the RCS barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability, and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within 2 hours based on a projection of current safety system performance. The term "imminent" refers to the inability to reach final safety acceptance criteria before completing 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.

Reactor Coolant System
Judgment - Potential Loss

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

The "Judgment" classifications should not be used if an applicable EAL classification has already been determined as included in the EAL chart.

PNPS Basis Reference(s):

None

Primary Containment
RPV Level - Loss

Barrier: Primary Containment

Category: A. RPV Level

Degradation Threat: Loss

Threshold:

None

Primary Containment
RPV Level - Potential Loss

Barrier: Primary Containment

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

22. SAG entry required

NEI 99-01 Basis:

There is no loss threshold associated with this item.

The potential loss requirement for Primary Containment flooding indicates adequate core cooling cannot be established and maintained and that core melt is possible. Entry into Primary Containment flooding procedures is a logical escalation in response to the inability to maintain adequate core cooling.

The condition in this potential loss threshold represents a potential core melt sequence which, if not corrected, could lead to vessel failure and increased potential for containment failure. In conjunction with reactor vessel water level "loss" thresholds in the fuel clad and RCS barrier columns, this threshold will result in the declaration of a General Emergency - loss of two barriers and the potential loss of a third.

Primary Containment
RPV Level - Potential Loss

PNPS Basis:

Severe Accident Guidelines (SAGs) are entered from EOP-1, EOP-2, EOP-3, EOP-16, and EOP-26 when Primary Containment flooding is required. Primary Containment flooding is required when core cooling is severely challenged. These EOPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined. SAG entry is required when (ref. 1):

- RPV water level cannot be restored and maintained above -150 in. (MSCRWL) (ref. 1).
- RPV water level cannot be restored and maintained at or above -175 in. (elevation of the jet pump suction) and no core spray subsystem flow can be restored and maintained equal to or greater than 3,600 GPM (design core spray flow) (ref. 1).
- RPV water level cannot be determined and core damage is occurring (ref. 2, 3).

The above EOP conditions, if not restored and maintained, represent a potential core melt sequence which could lead to RPV failure and increased potential for containment failure.

This threshold is also a loss of the fuel clad barrier (FC Loss A.1). Since SAG entry occurs after core uncovery has occurred, a loss of the RCS barrier exists (RCS Loss A.7). SAG entry, therefore, represents a loss of two barriers and a potential loss of a third, which requires a General Emergency classification.

PNPS Basis Reference(s):

1. EOP-1, *"RPV Control"*
2. EOP-16, *"RPV Flooding"*
3. EOP-26, "*RPV Flooding, Failure-to-Scram"*
4. EOP-2, "*RPV Control, Failure-to-Scram*"
5. EOP-3, *"Primary Containment Control"*

Primary Containment
PC Pressure/Temp - Loss

Barrier: Primary Containment

Category: B. PC Pressure/Temperature

Degradation Threat: Loss

Threshold:

16. PC pressure rise followed by a rapid unexplained drop in PC pressure

NEI 99-01 Basis:

Rapid unexplained loss of pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure increase from a high energy line break indicates a loss of containment integrity. Primary Containment pressure should increase as a result of mass and energy release into containment from a LOCA. Thus, Primary Containment pressure not increasing under these conditions indicates a loss of containment integrity.

This indicator relies on operator recognition of an unexpected response for the condition and therefore does not have a specific value associated with it. The unexpected response is important because it is the indicator for a containment bypass condition.

PNPS Basis:

None

PNPS Basis Reference(s):

None

Primary Containment
PC Pressure/Temp - Loss

Barrier: Primary Containment

Category: B. PC Pressure/Temperature

Degradation Threat: Loss

Threshold:

17. PC pressure response **not** consistent with LOCA conditions

NEI 99-01 Basis:

Primary Containment pressure should increase as a result of mass and energy release into containment from a LOCA.

This indicator relies on operator recognition of an unexpected response for the condition and therefore does not have a specific value associated with it. The unexpected response is important because it is the indicator for a containment bypass condition.

PNPS Basis:

The calculated pressure and temperature responses of the Primary Containment are shown in FSAR Figures 5.2-1 through 5.2-6 (ref. 1, 2). These figures show that the maximum calculated drywell pressure is well below the design allowable pressure (ref. 1, 2, 3). Due to conservatisms in LOCA analyses, actual pressure response is expected to be less than the analyzed response (ref. 2).

PNPS Basis Reference(s):

1. FSAR Figures 5.2-1 through 5.2-6
2. FSAR Table 5.2-1
3. PNPS-NE-07-00006 Rev. 0

Primary Containment
PC Pressure/Temp - Potential Loss

Barrier: Primary Containment

Category: B. PC Pressure/Temperature

Degradation Threat: Potential Loss

Threshold:

23. Torus bottom pressure > 60 psig and rising

NEI 99-01 Basis:

The potential loss pressure is based on the Primary Containment design pressure.

PNPS Basis:

When torus bottom pressure reaches the maximum allowable value (60 psig) (ref. 1), Primary Containment venting may be required even if offsite radioactivity release rate limits will be exceeded (ref. 2). The torus bottom pressure value of 60 psig is based on the Primary Containment design pressure as demonstrated in the PNPS accident analysis (ref. 1). If this threshold is exceeded, a challenge to the containment structure has occurred because assumptions used in the accident analysis are no longer valid and an unanalyzed condition exists. This constitutes a potential loss of the Primary Containment barrier even if a containment breach has not occurred.

PNPS Basis Reference(s):

1. FSAR Table 5.2-1
2. EOP-3, *"Primary Containment Control"*

Primary Containment
PC Pressure/Temp - Potential Loss

Barrier: Primary Containment

Category: B. PC Pressure/Temperature

Degradation Threat: Potential Loss

Threshold:

24. Deflagration concentrations exist inside PC

NEI 99-01 Basis:

None

PNPS Basis:

Deflagration (explosive) mixtures in the Primary Containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/SAMGs indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to Primary Containment integrity. Hydrogen concentration of approximately 6% is considered the global deflagration concentration limit (ref. 1).

Primary Containment
PC Pressure/Temp - Potential Loss

Except for brief periods during plant startup and shutdown, oxygen concentration in the Primary Containment is maintained at insignificant levels by nitrogen inertion. The specified values for this potential loss threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen, ref. 1) and readily recognizable because 6% hydrogen is well above the EOP-3, *"Primary Containment Control"*, entry condition (ref. 2). The minimum global deflagration hydrogen/oxygen concentrations (6% and 5%, respectively) require intentional Primary Containment venting, which is defined to be a loss of Primary Containment (PC Loss C.19).

The H2/O2 system monitors Primary Containment hydrogen (H2) and oxygen (O2) gas concentrations via the following containment penetration sample points: X-29E, X-106A-b, X-228J, X-15E, X-50A-d, and X-228C. The H2/O2 system returns its samples to Primary Containment via containment penetrations X-46F and X-228K.

The analyzer system consists of two separate units: an analyzer panel and a remote panel. Analyzer Panels C172 and C173 are located close to Primary Containment on the 74'3" elevation of the Reactor Building. Each panel contains all essential components necessary to perform the required containment atmosphere analysis, indication, and alarm functions. Recorders AR-1001-612A and AR-1001-612B and indicating meters are located in the Control Room in Panels C174 and C175. The backup Comsip H2/O2 analyzers, C172 and C173, are set to alarm at either 4% O2 or 4% H2 and are normally left in the "STANDBY" mode (ref. 3, 4, 5).

PNPS Basis Reference(s):

1. BWROG EPG/SAG Revision 2, Sections PC/G
2. EOP-3, *"Primary Containment Control"*
3. PNPS 2.2.120, *"Postaccident Monitoring Panel"*
4. PNPS 2.2.133, *"H2/O2 Analyzer and C19 Systems"*
5. SAG-02, *"Containment and Radioactivity Release Control"*

Barrier: Primary Containment

Category: B. PC Pressure/Temperature

Degradation Threat: Potential Loss

Threshold:

25. Torus water temperature and RPV pressure **cannot** be maintained below Heat Capacity Temperature Limit (EOP-11 Figure 2)

NEI 99-01 Basis:

The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool temperature from which emergency RPV depressurization will not raise:

- Suppression chamber temperature above the maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized;

OR

- Suppression chamber pressure above Primary Containment Pressure Limit A while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

The HCTL is a function of RPV pressure and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant and, therefore, the inability to maintain plant parameters below the limit constitutes a potential loss of containment.

Primary Containment
PC Pressure/Temp - Potential Loss

PNPS Basis:

The Heat Capacity Temperature Limit (HCTL) is the highest torus temperature from which emergency RPV depressurization will not raise torus pressure above the Primary Containment Pressure Limit (PCPL) while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent. The HCTL is a function of RPV pressure and torus level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant and, therefore, the inability to maintain plant parameters below the limit constitutes a potential loss of the Primary Containment barrier. This threshold is met when EOP-3, *"Primary Containment Control"*, Step TT-10 is reached (ref. 1).

PNPS Basis Reference(s):

1. EOP-3, *"Primary Containment Control"*

Primary Containment Isolation - Loss

Barrier: Primary Containment

Category: C. Isolation

Degradation Threat: Loss

Threshold:

18. Failure of **any** valve in **any** one line to close

    AND

    Direct downstream pathway to the environment exists after PC isolation signal

NEI 99-01 Basis:

This threshold addresses incomplete containment isolation that allows direct release to the environment.

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 product 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.

PNPS Basis:

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway directly to the environment. The concern is the unisolable open pathway to the environment. A failure of the ability to isolate any one line indicates a breach of Primary Containment integrity.

The adjective "direct" modifies "downstream pathway" to discriminate against release paths through interfacing liquid systems. Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include unisolable main steam line, HPCI steam line, or RCIC steam line breaks; unisolable RWCU system breaks; and unisolable containment atmosphere vent paths. If the main condenser is available with an unisolable main steam line, there may be releases through the steam jet air ejectors and gland seal exhausters. These pathways are monitored, however, and do not meet the intent of a nonisolable release path to the environment. These minor releases are assessed using the Category A, Abnormal Rad Release/Rad Effluent, EALs.

The existence of an in-line charcoal filter (SGTS) 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. Since Secondary Containment is not one of the three EAL fission product barriers, a direct unisolable release into Secondary Containment should therefore be considered a downstream pathway to the environment.

The threshold is met if the breach is not isolable from the Control Room or an attempt for isolation from the Control Room has been made and was unsuccessful. An attempt for isolation from the Control Room should be made prior to the emergency classification. If operator actions from the Control Room are successful, this threshold is not applicable. Credit is not given for operator actions taken in-plant (outside the Control Room) to isolate the breach.

EOP-3, *"Primary Containment Control"*, Step P-7 may specify Primary Containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1). Under these conditions, with a valid containment isolation signal, the Primary Containment barrier should be considered lost.

PNPS Basis Reference(s):

1. EOP-3, *"Primary Containment Control"*

Primary Containment Isolation - Loss

Barrier: Primary Containment

Category: C. Isolation

Degradation Threat: Loss

Threshold:

19. Intentional PC venting per EOPs

NEI 99-01 Basis:

The EOPs may direct containment isolation valve logic(s) to be intentionally bypassed regardless of radioactivity release rates. Under these conditions, with a valid containment isolation signal, the containment should also be considered lost if containment venting is actually performed.

Intentional venting of Primary Containment for Primary Containment pressure or combustible gas control in accordance with EOPs to the Secondary Containment and/or the environment is considered a loss of containment. Containment venting for pressure when not in an accident situation should not be considered.

PNPS Basis:

EOP-3, *"Primary Containment Control"*, Step P-7 may specify Primary Containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1). The threshold is met when the operator begins venting the Primary Containment in accordance with EOP-3, not when actions are taken to bypass interlocks prior to opening the vent valves. Purge and vent actions specified in EOP-3 Step P-1 to control drywell pressure below the drywell high pressure scram setpoint do not meet this threshold because such action is only permitted if offsite radioactivity release rates will remain below Technical Specifications LCO limits.

Primary Containment Isolation - Loss

PNPS Basis Reference(s):

1. EOP-3, *"Primary Containment Control"*

Primary Containment Isolation - Loss

Barrier: Primary Containment

Category: C. Isolation

Degradation Threat: Loss

Threshold:

20. Unisolable primary system discharge outside PC resulting in Secondary Containment area radiation or temperature above **any** Maximum Safe Operating Value (EOP-4, Table L)

NEI 99-01 Basis:

This loss threshold addresses the presence of area radiation or temperature Max Safe Operating Values indicating unisolable primary system leakage outside the Primary Containment after a containment isolation. The indicators should be confirmed to be caused by RCS leakage.

There is no potential loss threshold associated with this item.

PNPS Basis:

The Maximum Safe Operating Values define this Primary Containment barrier threshold because they are indicative of problems in the Secondary Containment that are spreading and pose a threat to achieving a safe plant shutdown. This threshold addresses problematic discharges outside Primary Containment that may not originate from a high-energy line break. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-4, Secondary Containment Control, Table L (ref. 1) (Table F-3 below).

Primary Containment Isolation - Loss

In general, multiple indications should be used to determine whether a primary system is discharging outside Primary Containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the Secondary Containment since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g., room flooding, high area temperatures, reports of steam in the Secondary Containment, an unexpected rise in feedwater flow rate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the Secondary Containment.

PNPS Basis Reference(s):

1. EOP-4, *"Secondary Containment Control"*

Primary Containment Isolation - Loss

Table F-3 Secondary Containment Area Temperature and Radiation Maximum Safe Operating Values

RADIATION LEVEL		VALUE (mR/hr)
NW equipment space/HPCI pump room - 17 ft 6 in. El.		1000
CRD pump room - 17 ft 6 in. El.		1000
RCIC pump room - 17 ft 6 in. El.		1000
SE equipment space - 17 ft 6 in. El.		1000
CRD HCU west area - 23 ft El.		1000
CRD HCU east area - 23 ft El.		1000
RB west area - 51 ft El.		1000
RB east area - 51 ft El.		1000
North Storage and laydown area - 74 ft 3 in. El. (H202)		1000
Fuel pool cooling pump/HX area - 74 ft 3 in. El		1000
SLC pump area - 91 ft 3 in. El.		1000
Skimmer surge tank area - 91 ft 3 in. El.		1000
TEMPERATURE		**VALUE (°F)**
(Temperature areas are separated by dashed lines)RWCU Holding Pump Area - 74 ft El.	TE-1291-60C	120
RWCU Filter Area - 74 ft El.	TE-1291-60A	120
RWCU Backwash Tank Area - 51 ft El.	TE-1291-60B	214
RWCU Pump "A" Room - 51 ft El.	TE-1291-60D	213
RWCU Pump "B" Room - 51 ft El.	TE-1291-60E	213
RWCU Heat Exchanger Room - 51 ft El.	TE-1291-60F	215
RCIC Piping Area - Torus Compt	TE-1360-23A	258
RCIC Turbine Area - Stairwell	TE-1360-23B	175
HPCI Piping Area - Torus Compt	TE-2374A	258
HPCI Turbine Area - 17 ft El.	TE-2374B	175
RWCU Piping Area - 36 ft El. Mezzanine	TE-1291-60H	238
RCIC Tip Room - 23 ft El.	TE-1360-23C	224
Main Steam Tunnel - 23 ft El.	TE-261-22A	289
HPCI Piping Area - 23 ft El. ("B" RHR Valve Room)	TE-2374C	309
RHR "B" & "D" Pump Area - Stairwell	TE-1001-92A	200
RHR "A" & "C" Pump Area - 6 ft El.	TE-1001-92B	200
RHR "A" & "C" Pump Area - Pipewell	TE-1001-92G	224
RWCU & RHR Valve Room - 23 ft El. ("A" RHR Vlv Rm)	TE-1001-92F	251
RHR Fuel Pool Heat Exchanger Room - 74 ft El.	TE-1001-92H	120

Primary Containment Isolation - Potential Loss

Barrier: Primary Containment

Category: C. Isolation

Degradation Threat: Potential Loss

Threshold:

None

Primary Containment
Rad - Loss

Barrier: Primary Containment

Category: D. Rad

Degradation Threat: Loss

Threshold:

None

Primary Containment
Rad - Potential Loss

Barrier: Primary Containment

Category: D. Rad

Degradation Threat: Potential Loss

Threshold:

26. Drywell High Range Area Radiation Monitor (RIT-1001-606A and B) > 8,000 R/hr

    OR

    Torus High Range Area Radiation Monitor (RIT-1001-607A and B) > 500 R/hr

NEI 99-01 Basis:

The potential loss threshold reading is a value that indicates significant fuel damage well in excess of that required for loss of RCS and fuel clad. 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%.

There is no loss threshold associated with this item.

Primary Containment
Rad - Potential Loss

PNPS Basis:

The Drywell and Torus High Range Area Radiation Monitor readings indicate the release of reactor coolant into the drywell with elevated activity indicative of fuel damage. The reading was derived assuming (ref. 1):

- The reactor has been shutdown for 1 hour
- 20% fuel clad damage
- No drywell sprays in operation
- A LOCA-depressurized system
- The instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory into the drywell atmosphere

The monitor reading of 1 hour after shutdown provides a realistic time for a degraded fuel condition to develop in a fast-breaking accident, which results in conservative threshold values for releases to containment with less than 1 hour after shutdown (ref. 1).

In order to reach this Primary Containment barrier potential loss threshold, a loss of the RCS barrier (RCS Loss D.11) and a loss of the fuel clad barrier (FC Loss D.2) have already occurred. This threshold, therefore, represents at a General Emergency classification.

PNPS Basis Reference(s):

1. EP-IP-330, *"Core Damage"*

Primary Containment
Judgment - Loss

Barrier: Primary Containment

Category: E. Judgment

Degradation Threat: Loss

Threshold:

21. **Any** condition in the opinion of the Emergency Director that indicates loss of the PC barrier

NEI 99-01 Basis:

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the containment barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

The containment barrier should not be declared lost based on exceeding Technical Specifications action statement criteria unless there is an event in progress requiring mitigation by the containment barrier. When no event is in progress (loss or potential loss of either fuel clad and/or RCS), the containment barrier status is addressed by Technical Specifications.

Primary Containment
Judgment - Loss

PNPS Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining whether the Primary Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability, and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within 2 hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

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

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

The "Judgment" classifications should not be used if an applicable EAL classification has already been determined as included in the EAL chart.

PNPS Basis Reference(s):

None

Primary Containment
Judgment - Potential Loss

Barrier: Primary Containment

Category: E. Judgment

Degradation Threat: Potential Loss

Threshold:

27. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the PC barrier

NEI 99-01 Basis:

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the containment barrier is potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

The containment barrier should not be declared potentially lost based on exceeding Technical Specifications action statement criteria unless there is an event in progress requiring mitigation by the containment barrier. When no event is in progress (loss or potential loss of either fuel clad and/or RCS), the containment barrier status is addressed by Technical Specifications.

Primary Containment
Judgment - Potential Loss

PNPS Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining whether the Primary Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability, and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within 2 hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

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

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

The "Judgment" classifications should not be used if an applicable EAL classification has already been determined as included in the EAL chart.

PNPS Basis Reference(s):

None

Unusual Event - EU1.1

Category: E - ISFSI

Subcategory: None

Initiating Condition: Occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated.

EAL:

EU1.1 Unusual Event

Damage to a loaded cask CONFINEMENT BOUNDARY.

Mode Applicability:

All

NEI 99-01 Basis:

A NOUE 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.

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 identify natural phenomena events and accident conditions that could potentially effect the CONFINEMENT BOUNDARY. This EAL addresses a dropped cask, a tipped over cask, EXPLOSION, PROJECTILE damage, FIRE damage or natural phenomena affecting a cask (e.g., seismic event, tornado, etc.).

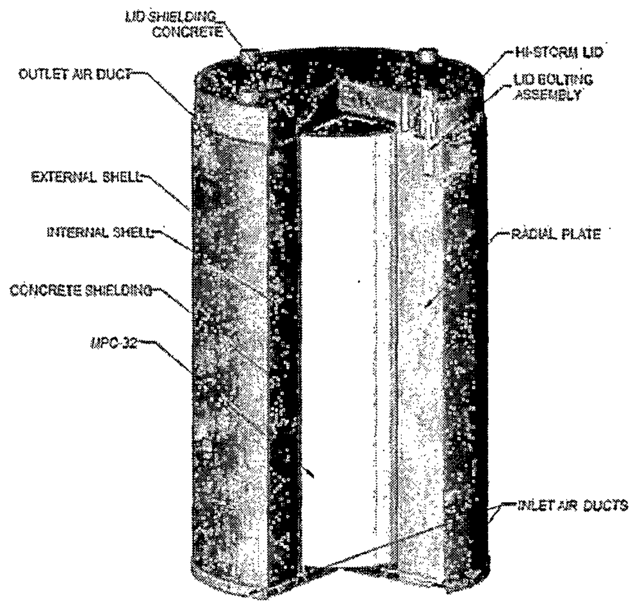
Unusual Event - EU1.1

PNPS Basis:

An Unusual Event is categorized as an event of sufficient magnitude that the CONFINEMENT BOUNDARY of a cask loaded with spent fuel is damaged such that it is breached.

Confinement Boundary means the outline formed by the sealed, cylindrical enclosure of the Multi-Purpose Canister (MPC) shell welded to a solid baseplate, a lid welded around the top circumference of the shell wall, the port cover plates welded to the lid, and the closure ring welded to the lid and MPC shell providing the redundant sealing.

HI-STORM 100 Overpack



This diagram is also applicable for the MPC-68

PNPS Basis Reference(s):

1. Holtec International FSAR for the Hi-Storm 100 Cask System revision 9
2. EC 53194 - Dry Fuel Project - Design Basis Threat Evaluation (DBT Blast Effects on Loaded HI-Storm)
3. EC28039 - Dry Fuel Storage Operations

9.8.1 BACKGROUND

[1] EALs are the plant-specific indications, conditions, or instrument readings that are utilized to classify emergency conditions defined in the PNPS Emergency Plan (ref. 2.1[6].

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 5 represents the most recently accepted methodology. Enhancements over earlier revisions include:

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

(b) 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 (ISFSIs).

(c) Simplifying the fission product barrier EAL threshold for a Site Area Emergency.

[2] Using NEI 99-01 Revision 5, Pilgrim Station conducted an EAL implementation upgrade project that produced the EALs discussed herein.

9.8.2 FISSION PRODUCT BARRIERS

[1] 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 or potential loss of 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; "potential loss" infers an increased probability of barrier loss and decreased certainty of maintaining the barrier.

[2] The primary fission product barriers are:

(a) A - <u>Fuel Clad (FC):</u> The fuel clad barrier consists of the zircalloy fuel bundle tubes that contain the fuel pellets.

(b) B - <u>Reactor Coolant System (RCS):</u> 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.

(c) C - <u>Primary Containment (PC):</u> The Primary Containment barrier includes the drywell, the wetwell (torus), their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves.

9.8.3 EMERGENCY CLASSIFICATION BASED ON FISSION PRODUCT BARRIER DEGRADATION

[1] The following criteria are the bases for event classification related to fission product barrier loss or potential loss:

(a) *<u>Unusual Event:</u>*

Any loss or any potential loss of Primary Containment

(b) *<u>Alert:</u>*

Any loss or any potential loss of either fuel clad or RCS

(c) *<u>Site Area Emergency:</u>*

Loss or potential loss of any two barriers

(d) *<u>General Emergency:</u>*

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

9.8.4 EAL RELATIONSHIP TO EOPS

[1] Where possible, the EALs have been made consistent with and utilize the conditions defined in the PNPS Emergency Operating Procedures (EOPs). While the symptoms that drive operator actions specified in the EOPs are not indicative of all possible conditions which warrant emergency classification, they define the symptoms, independent of initiating events, for which reactor plant safety and/or fission product barrier integrity are threatened. When 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.

9.8.5 SYMPTOM-BASED VS. EVENT-BASED APPROACH

[1] To the extent possible, the EALs are symptom-based. That is, the action level threshold 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. However, 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.

9.8.6 EAL ORGANIZATION

[1] The PNPS EAL scheme includes the following features:

(a) Division of the EAL set into three broad groups:

(1) EALs applicable under all plant operating modes - This group would be reviewed by the EAL user any time emergency classification is considered.

(2) EALs applicable only under hot operating modes - This group would only be reviewed by the EAL user when the plant is in Hot Shutdown, Startup, or Run mode.

(3) EALs applicable only under cold operating modes - This group would only be reviewed by the EAL user when the plant is in Cold Shutdown, Refuel, or Defueled mode.

(b) The purpose of the groups is to avoid review of hot condition EALs when the plant is in a cold condition and avoid review of cold condition EALs when the plant is in a hot condition. This approach significantly minimizes the total number of EALs that must be reviewed by the EAL user for a given plant condition, reduces EAL user reading burden, and thereby speeds identification of the EAL that applies to the emergency.

(c) Within each of the three EAL groups described above, assignment of EALs to categories/subcategories - category and subcategory titles are selected to represent conditions that are operationally significant to the EAL user. Subcategories are used as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The proposed PNPS EAL categories/subcategories and their relationship to NEI Recognition Categories are listed below.

(d) The primary tool for determining the emergency classification level is the EAL wall chart. The user of the EAL wall chart may (but is not required to) consult the EAL Technical Bases in order to obtain additional information concerning the EALs under classification consideration. The user should consult Sections 9.8.7 and 9.8.8 and Attachments 9.1 through 9.7 of this document for such information.

(e) EAL Groups, Categories, and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
A - Abnormal Rad Release/Rad Effluent	1 - Offsite Rad Conditions 2 - Onsite Rad Conditions & Spent Fuel Pool Events 3 - MCR/CAC Radiation
H - Hazards	1 - Natural or Destructive Phenomena 2 - Fire or Explosion 3 - Hazardous Gas 4 - Security 5 - Control Room Evacuation 6 - Judgment
E - IFSFI	None
Hot Conditions:	
S - System Malfunction	1 - Loss of AC Power 2 - ATWS/Criticality 3 - Inability to Reach Shutdown Conditions 4 - Instrumentation/Communications 5 - Fuel Clad Degradation 6 - RCS Leakage 7 - Loss of DC Power
F - Fission Product Barrier Degradation	None
Cold Conditions:	
C - Cold Shutdown/Refuel System Malfunction	1 - Loss of AC Power 2 - RPV Level 3 - RCS Temperature 4 - Communications 5 - Inadvertent Criticality 6 - Loss of DC Power

9.8.7 TECHNICAL BASES INFORMATION

[1] EAL technical bases are provided in Attachment 9.1 through 9.7 for each EAL according to EAL group (Any, Hot, Cold), EAL category (A, H, E, S, F, and C), and EAL subcategory. A summary explanation of each category and subcategory is given at the beginning of the technical bases discussions of the EALs included in the category. For each EAL, the following information is provided:

(a) Category Letter & Title

(b) Subcategory Number & Title

(c) Initiating Condition (IC)

(d) Site-specific description of the generic IC given in NEI 99-01

[2] EAL Identifier (enclosed in rectangle)

(a) Each EAL is assigned a unique identifier to support accurate communication of the emergency classification to onsite and offsite personnel. Four characters define each EAL identifier:

(1) First character (letter): Corresponds to the EAL category as described above (A, H, E, S, F, or C).

(2) Second character (letter): The emergency classification (G, S, A, or U).

(3) Third character (number): Subcategory number within the given category. Subcategories are sequentially numbered beginning with the number one (1). If a category does not have a subcategory, this character is assigned the number one (1).

(4) Fourth character (number): The numerical sequence of the EAL within the EAL subcategory. If the subcategory has only one EAL, it is given the number one (1).

[3] Classification (enclosed in rectangle). These emergency classifications include:

(a) Unusual Event (U)

(b) Alert (A)

(c) Site Area Emergency (S)

(d) General Emergency (G)

[4] EAL (enclosed in rectangle)

(a) Exact wording of the EAL as it appears in the EAL classification matrix.

[5] Mode Applicability

(a) One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Run, 2 - Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refuel, D - Defueled, All, or N/A - Not Applicable. (See Section 9.8.8 for operating mode definitions.)

[6] Basis:

(a) A generic basis section provides a description of the rationale for the EAL as provided in NEI 99-01. This is followed by a plant-specific basis section that provides PNPS-relevant information concerning the EAL.

[7] PNPS Basis Reference(s):

(a) Site-specific source documentation from which the EAL is derived.

9.8.8 OPERATING MODE APPLICABILITY

[1] There are six Operating Modes, as follows:

(a) Mode 1: Run

(1) Reactor is critical and the mode switch is in RUN. In this mode, the reactor system pressure is at or above 785 psig and the Reactor Protection System is energized with APRM protection and RBM interlocks in service.

(b) Mode 2: Startup

(1) The mode switch is in STARTUP. In this mode the reactor protection scram trip, initiated by main steam line isolation valve closure, is bypassed when reactor pressure is less than 600 psig, the low pressure main steam line isolation valve closure trip is bypassed, the Reactor Protection System is energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service.

(c) Mode 3: Hot Shutdown

(1) The mode switch is in SHUTDOWN, no core alterations are being performed, and reactor coolant temperature is > 212°F.

(d) Mode 4: Cold Shutdown

(1) The mode switch is in SHUTDOWN, no core alterations are being performed, and reactor coolant temperature is ≤ 212°F.

(e) Mode 5: Refuel

(1) The mode switch is in REFUEL and reactor coolant temperature is ≤ 212°F. During normal reactor shutdowns, the mode switch may be placed in the REFUEL position to proceed with manual control rod insertion. Although the mode switch is in REFUEL position for this evolution, refuel mode applicability for the purpose of EAL classification is defined by reactor coolant temperature as well as mode switch position.

(f) Mode DEF: Defueled

(1) Reactor vessel contains no irradiated fuel (full core off-load during refueling or extended outage).

[2] 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.

[3] For events that occur in cold shutdown or refueling, escalation is via EALs that have cold shutdown or refueling for mode applicability, even if hot shutdown (or a higher mode) is entered during any subsequent heat-up. In particular, the fission product barrier EALs are applicable only to events that initiate in hot shutdown or higher.

9.8.9 VALIDATION OF INDICATIONS, REPORTS, AND CONDITIONS

[1] All emergency classifications shall be based upon valid indications, reports, or conditions. 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's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

9.8.10 PLANNED VS. UNPLANNED EVENTS

[1] Planned evolutions involve preplanning to address the limitations imposed by the condition, the performance of required surveillance testing, and the implementation of specific controls prior to knowingly entering the condition in accordance with the specific requirements of the site's Technical Specifications. Activities which cause the site to operate beyond that allowed by the site's Technical Specifications, planned or unplanned, may result in an EAL threshold being met or exceeded. Planned evolutions to test, manipulate, repair, perform maintenance or modifications to systems and equipment that result in an EAL value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned and is within the operational limitations imposed by the specific operating license. However, these conditions may be subject to the reporting requirements of 10CFR50.72.

9.8.11 CLASSIFYING TRANSIENT EVENTS

[1] For some events the condition may be corrected before a declaration has been made. 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 radiochemistry sampling) may be necessary. 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.

[2] Existing guidance for classifying transient events addresses the period of time of event recognition and classification (15 minutes). However, in cases when EAL declaration criteria may be met momentarily during the normal expected response of the plant, declaration requirements should not be considered to be met when the conditions are a part of the designed plant response or result from appropriate operator actions.

[3] There may be cases in which a plant condition that exceeded an EAL 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.

[4] Reporting requirements of 10CFR50.72 are applicable and the guidance of NUREG-1022, Event Reporting Guidelines 10CFR50.72 and 50.73, should be applied.

9.8.12 IMMINENT EAL THRESHOLDS

[1] 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 (the early classification may permit more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.

9.8.13 TREATMENT OF MULTIPLE EVENTS

[1] When multiple simultaneous events occur, the emergency classification level is based on the highest EAL reached. For example: two Alerts remain in the Alert category; an Alert and a Site Area Emergency is a Site Area Emergency.

AC ....Alternating Current

ADS ....Automatic Depressurization System

APRM ....Average Power Range Monitor

ARI ....Alternate Rod Insertion associated instrumentation

ATWS ....Anticipated Transient Without Scram

BIIT ....Boron Injection Initiation Temp

BWR ....Boiling Water Reactor

CAC ....Containment Atmospheric Control

CAD ....Containment Atmospheric Dilution

CDE ....Committed Dose Equivalent

CFR ....Code of Federal Regulations

cps ....Counts per Second

CRD ....Control Rod Drive

CS ....Core Spray

CSCS ....Core Standby Cooling System

CST ....Condensate Storage Tank

DC ....Direct Current

Demin ....Demineralizer

DHRP ....Decay Heat Removal Pressure

DW ....Drywell

DWSIL ....Drywell Spray Initiation Limit

EAL ....Emergency Action Level

ECCS ....Emergency Core Cooling Systems

ECL ....Emergency Classification Level

ED ....Emergency Director

El. ....Elevation

EOF....................Emergency Operations Facility

EOP....................Emergency Operating Procedure

EPA....................Environmental Protection Agency

EPG....................Emergency Procedure Guideline

EPIP..................Emergency Plan Implementing Procedure

EPRI..................Electric Power Research Institute

ESF....................Engineered Safety Feature

FAA....................Federal Aviation Administration

FBI......................Federal Bureau of Investigation

FEMA................Federal Emergency Management Agency

FSAR.................Final Safety Analysis Report

ft.........................Feet

FW......................Feedwater

gal......................Gallon(s)

GE......................General Emergency

GPM..................Gallons Per Minute

H&V....................Heating and Ventilation

H2......................Hydrogen

HCTL.................Heat Capacity Temperature Limit

HCU...................Hydraulic Control Unit

HOO..................Headquarters (NRC) Operations Officer

HPCI..................High Pressure Coolant Injection

hr........................Hour

HX......................Heat Exchanger

IC........................Initiating Condition

IDLH..................Immediately Dangerous to Life and Health

in........................Inch(es)

IPEEE................Individual Plant Examination of External Events (Generic Letter 88-20)

ISFSI ....Independent Spent Fuel Storage Installation

Keff....................Effective Neutron Multiplication Factor

lb........................Pound(s)

LCO ....Limiting Condition for Operation

LER ....Licensee Event Report

LFL ....Lower Flammability Limit

LI ....Level Indicator

LOCA.................Loss of Coolant Accident

LPCI ....Low Pressure Coolant Injection

LPSI...................Low Pressure Safety Injection

LR......................Level Recorder

LWR ....Light Water Reactor

MDRIR...............Minimum Debris Retention Injection Rate

MDSL ....Minimum Debris Submergence Level

min.....................Minimum

MNSDHR...........Minimum Number of SRVs Required for Decay Heat Removal

MNSRED...........Minimum Number of SRVs Required for Emergency Depressurization

mR.....................Milliroentgen/Millirem (as appropriate to the context and the units of radiation dose measurement)

mrem ....Millirem

MSCP ....Minimum Steam Cooling Pressure

MSIV..................Main Steam Isolation Valve

MSL...................Main Steam Line

MW....................Megawatt

NA ....Not applicable
NEI ....Nuclear Energy Institute
NESP.................National Environmental Studies Project
NORAD ....North American Aerospace Defense Command
NOUE ....Notification Of Unusual Event
NPP ....Nuclear Power Plant
NPSH ....Net positive suction head
NR ....Narrow Range
NRC...................Nuclear Regulatory Commission
NSSS.................Nuclear Steam Supply System
NW ....Northwest
O2......................Oxygen
OBE...................Operating Basis Earthquake
OCA...................Owner Controlled Area
ODCM/ODAM....Offsite Dose Calculation (Assessment) Manual
PA......................Protected Area
PAG...................Protective Action Guideline
PASS.................Postaccident Sample System
PC ....Primary Containment
PCPL.................Primary Containment Pressure Limit
POAH ....Point of Adding Heat
PRA/PSA...........Probabilistic Risk Assessment/Probabilistic Safety Assessment
PRM ....Process Radiation Monitor
psig....................Pounds per square inch (gauge)
PSP ....Pressure Suppression Pressure
PSTG.................Plant Specific Technical Guidelines
PWR..................Pressurized Water Reactor

R........................Roentgen
RB......................Reactor Building
RBCCW.............Reactor Building Closed Cooling Water
RBVE.................Reactor Building Ventilation Exhaust
RCIC..................Reactor Core Isolation Cooling
RCS...................Reactor Coolant System
rem.....................Roentgen Equivalent Man
REQ'D...............Required
RHR...................Residual Heat Removal
RPM..................Revolutions per minute
RPS...................Reactor Protection System
RPV...................Reactor Pressure Vessel
RWCU...............Reactor Water Cleanup
SAG...................Severe Accident Guideline
SATG.................Severe Accident Technical Guideline
SBGTS..............Stand-By Gas Treatment System
SBO...................Station Blackout
SE......................Southeast
SGT...................Standby Gas Treatment
SLC...................Standby Liquid Control
SPDS.................Safety Parameter Display System
SRO...................Senior Reactor Operator
SRV...................Safety Relief Valve
SSE...................Safe Shutdown Earthquake
SSW..................Salt Service Water
SW....................Southwest

TAF....................Top of Active Fuel

TBCCW.............Turbine Building Closed Cooling Water

TEDE.................Total Effective Dose Equivalent

TI.......................Temperature Indicator

TIP.....................Traversing In-Core Probe

TPLL..................SRV Tailpipe Level Limit

TSC...................Technical Support Center

WB....................Whole Body

WR....................Wide Range

'..........................Feet

"..........................Inches

%........................Percent

&........................Ampersand ("and")

°F.......................Degrees Fahrenheit

>........................Greater Than

<........................Less Than

≥........................Greater Than or Equal To

≤........................Less Than or Equal To

This cross-reference is provided to facilitate association and location of a PNPS EAL within the NEI 99-01 IC/EAL identification scheme. Further information regarding the development of the PNPS EALs based on the NEI guidance can be found in the EAL Comparison Matrix.

PNPS	NEI 99-01	
EAL	IC	Example EAL
AU1.1	AU1	1
AU1.2	AU1	1
AU1.3	AU1	3
AU2.1	AU2	1
AU2.2	AU2	2
AA1.1	AA1	1
AA1.2	AA1	1
AA1.3	AA1	3
AA2.1	AA2	2
AA2.2	AA2	1
AA3.1	AA3	1
AS1.1	AS1	1
AS1.2	AS1	2
AS1.3	AS1	4
AG1.1	AG1	1
AG1.2	AG1	2
AG1.3	AG1	4

ATTACHMENT 9.10 **PNPS-To-NEI 99-01 EAL Cross-Reference**

Sheet 2 of 5

PNPS	NEI 99-01	
EAL	**IC**	**Example EAL**
CU1.1	CU3	1
CU6.1	CU7	1
CU2.1	CU1	1
CU2.2	CU2	1
CU2.3	CU2	2
CU3.1	CU4	1
CU3.2	CU4	2
CU4.1	CU6	1, 2
CU5.1	CU8	1
CA1.1	CA3	1
CA2.1	CA1	1, 2
CA3.1	CA4	1, 2
CS2.1	CS1	1
CS2.2	CS1	2
CS2.3	CS1	3
CG2.1	CG1	1
CG2.2	CG1	2
EU1.1	E-HU1	1
FU1.1	FU1	1
FA1.1	FA1	1

PNPS	**NEI 99-01**	
EAL	**IC**	**Example EAL**
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1
HU1.2	HU1	2
HU1.3	HU1	4
HU1.4	HU1	3
HU1.5	HU1	5
HU2.1	HU2	1
HU2.2	HU2	2
HU3.1	HU3	1
HU3.2	HU3	2
HU4.1	HU4	1, 2, 3
HU6.1	HU5	1
HA1.1	HA1	1
HA1.2	HA1	2
HA1.3	HA1	5
HA1.4	HA1	4
HA1.5	HA1	3
HA1.6	HA1	6

PNPS	NEI 99-01	
EAL	IC	Example EAL
HA2.1	HA2	1
HA3.1	HA3	1
HA4.1	HA4	1, 2
HA5.1	HA5	1
HA6.1	HA6	1
HS4.1	HS4	1
HS5.1	HS2	1
HS6.1	HS3	1
HG4.1	HG1	1, 2
HG6.1	HG2	1
SU1.1	SU1	1
SU2.1	SU8	1
SU3.1	SU2	1
SU4.1	SU3	1
SU4.2	SU6	1, 2
SU5.1	SU4	1
SU5.2	SU4	2
SU6.1	SU5	1, 2
SA1.1	SA5	1
SA2.1	SA2	1

ATTACHMENT 9.10 **PNPS-To-NEI 99-01 EAL Cross-Reference**

Sheet 5 of 5

PNPS	NEI 99-01	
EAL	**IC**	**Example EAL**
SA4.1	SA4	1
SS1.1	SS1	1
SS7.1	SS3	1
SS2.1	SS2	1
SS4.1	SS6	1
SG1.1	SG1	1
SG2.1	SG2	1

ATTACHMENT 9.11 **EAL PAGE LISTING**

Sheet 1 of 1

Category A - Abnormal Rad Release/Rad Effluent

EAL	Page #
AU1.1	18
AU1.2	23
AU1.3	27
AU2.1	55
AU2.2	58
AA1.1	30
AA1.2	34
AA1.3	38
AA2.1	60
AA2.2	63
AA3.1	66
AS1.1	40
AS1.2	44
AS1.3	46
AG1.1	48
AG1.2	51
AG1.3	53

Category S - System Malfunction

EAL	Page #
SU1.1	141
SU2.1	157
SU3.1	169
SU4.1	171
SU4.2	173
SU5.1	182
SU5.2	184
SU6.1	185
SA1.1	145
SA2.1	158
SA4.1	175
SS1.1	149
SS2.1	162
SS4.1	178
SS7.1	187
SG1.1	153
SG2.1	165

Category F - Fission Product Barrier Degradation

EAL	Page #
FU1.1	192
FA1.1	193
FS1.1	194
FG1.1	196
1	265
2	273
3	275
4	277
5	267
6	279
7	281
8	284
9	287
10	289
11	295
12	298
13	291
14	292
15	300
16	305
17	306
18	312
19	314
20	316
21	323
22	303
23	307
24	308
25	310
26	321
27	325

Category H - Hazards

HU1.1	70
HU1.2	72
HU1.3	74
HU1.4	76
HU1.5	78
HU2.1	98
HU2.2	101
HU3.1	106
HU3.2	109
HU4.1	114
HU6.1	130
HA1.1	81
HA1.2	84
HA1.3	87
HA1.4	89
HA1.5	92
HA1.6	95
HA2.1	103
HA3.1	111
HA4.1	118
HA5.1	127
HA6.1	132
HS4.1	122
HS5.1	128
HS6.1	134
HG4.1	125
HG6.1	136

Category C - Cold Shutdown/Refueling System Malfunction

CU1.1	200
CU2.1	208
CU2.2	211
CU2.3	215
CU3.1	245
CU3.2	248
CU4.1	256
CU5.1	258
CU6.1	259
CA1.1	204
CA2.1	219
CA3.1	252
CS2.1	224
CS2.2	226
CS2.3	228
CG2.1	232
CG2.2	237

Category E - ISFSI (Independent Spent Fuel Storage Installation)

EU1.1	327

Unusual Event

Alert

Site Area Emergency

General Emergency

ATTACHMENT 9.12 **DOCUMENT CROSS-REFERENCES**

Sheet 1 of 1

This Attachment lists those documents, other than source documents, which may be affected by changes to this Procedure.

Document Number	Document Title
EP-IP-100	Emergency Classification and Notification
EP-IP-100.1	Emergency Action Levels (EALs)
	EAL Wall Chart
EN-EP-313	Offsite Dose Assesment using the Unified RASCAL Interface
EP-IP-310	Offsite Monitoring Team Activation and Response
EP-IP-330	Core Damage
EP-IP-400	Protective Action Recommendations
PNPS 5.3.14	Security Incidents
PNPS 5.3.14.1	Airborne Threat

Procedure/Document Number: EP-AD-601	Revision: 6
Equipment/Facility/Other: Pilgrim Nuclear Power Station	
Title: Emergency Action Level Technical Bases Document	

Part I. Description of Activity Being Reviewed (event or action, or series of actions that may result in a change to the emergency plan or affect the implementation of the emergency plan):

The non-intent changes to this document revise the description of a "felt earthquake" under EAL HU1.1 by removing the words: "As defined in the EPRI-sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989," and also removing: " and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated", adding wording to several PNPS EAL Basis discussions as recommended by PNPS Operations personnel to reinforce existing guidance, and emboldening selected existing wording for the security EAL HU4.1 to reinforce Security's role in recognizing security based events. See the attached Revision Matrix Review of Changes.

Part II. Activity Previously Reviewed?		
Is this activity fully bounded by an NRC approved 10 CFR 50.90 submittal or Alert and Notification System Design Report? If YES, identify bounding source document number/approval reference and ensure the basis for concluding the source document fully bounds the proposed change is documented below:	☐ YES 50.54(q)(3) Evaluation is NOT required. Enter justification below and complete Part VI.	☒ NO Continue to next part

Justification:

☐ Bounding document attached (optional)

Part III. Applicability of Other Regulatory Change Control Processes

APPLICABILITY CONCLUSION

☒ If there are no controlling change processes, continue the 50.54(q)(3) Screening.

☐ One or more controlling change processes are selected, however, some portion of the activity involves the emergency plan or affects the implementation of the emergency plan; continue the 50.54(q)(3) Screening for that portion of the activity. Identify the applicable controlling change processes below.

☐ One or more controlling change processes are selected and fully bounds all aspects of the activity. 50.54(q)(3) Evaluation is NOT required. Identify controlling change processes below and complete Part VI.

CONTROLLING CHANGE PROCESSES

10 CFR 50.54(q)

Part IV. Editorial Change		
Part IV. Editorial Change Is this activity an editorial or typographical change such as formatting, paragraph numbering, spelling, or punctuation that does not change intent? Yes: Items 4 and 5. **Justification: Items 4 and 5 on the attached Revision Matrix Review of Changes embolden existing wording in the PNPS Basis to emphasize Security's role in recognizing security based events. This is an editorial change that does not change intent of the EALs.** "NO" is checked because the PNPS Emergency Plan revision contains other changes that are not editorial.	☐ YES 50.54(q)(3) Evaluation is NOT required. Enter justification and complete Part VI.	☒ NO Continue to next part

Procedure/Document Number: EP-AD-601	**Revision: 6**
Equipment/Facility/Other: Pilgrim Nuclear Power Station	
Title: Emergency Action Level Technical Basis Document	

21. The resources for controlling radiological exposures for emergency workers are established. [11]	☐
22. Arrangements are made for medical services for contaminated, injured individuals. [12]	☐
23. Plans for recovery and reentry are developed. [13]	☐
24. A drill and exercise program (including radiological, medical, health physics and other program areas) is established. [14]	☐
25. Drills, exercises, and training evolutions that provide performance opportunities to develop, maintain, and demonstrate key skills are assessed via a formal critique process in order to identify weaknesses. [14]	☐
26. Identified weaknesses are corrected. [14]	☐
27. Training is provided to emergency responders. [15]	☐
28. Responsibility for emergency plan development and review is established. [16]	☐
29. Planners responsible for emergency plan development and maintenance are properly trained. [16]	☐

APPLICABILITY CONCLUSION

X If no Part V criteria are checked, a 50.54(q)(3) Evaluation is NOT required; document the basis for conclusion below and complete Part VI.

☐ If any Part V criteria are checked, complete Part VI and perform a 50.54(q)(3) Evaluation.

BASIS FOR CONCLUSION

Changes numbered 1 through 3 and 6 through 9 on the attached Revision Matrix Review of Changes were developed as suggested by and in concert with Entergy Corporate and PNPS Operations personnel to reinforce existing guidance within the subject PNPS EAL Bases discussions. The Basis for Conclusion for each of these changes is included in detail within the attached Revision Matrix. These changes do not change intent, facilities, equipment or processes for this procedure or affect any planning standard elements. This activity does not affect the PNPS Emergency Plan. No further evaluation is required for this activity.

Part VI. Signatures:

Preparer Name (Print) Karen Sullivan	Preparer Signature ~~Electronic Signature~~	Date: 2/1/2016
(Optional) Reviewer Name (Print)	Reviewer Signature	Date:
Reviewer Name (Print) Jack Lewis Nuclear EP Project Manager	Reviewer Signature	Date: 2-2-16
Approver Name (Print) Donna Calabrese EP manager or designee	Approver Signature	Date: 2/3/2016

Revision Matrix for 50.54(q): Review of Changes

Revision 6 of EAL Technical Bases Document

Item #	Description of Change	Section	Editorial Change per Reg Guide 2.129?	Affect on 10 CFR 50.47(b) Planning Standards or NUREG 0654 program elements?	Planning standard affected	Basis for Conclusion
1	Revise description of earthquake by removing: "As defined in the EPRI-sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989,...", and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated."	EAL HU1.1, Page 70	No	No	N/A	NO - This is an administrative change that removes reference to the guidelines previously used to describe an "earthquake". Since the description is being changed (as discussed under Item 2 below), and will no longer represent the description as provided in this reference, the reference is no longer valid. This change does not change intent, facilities, equipment or processes for this procedure or affect any planning standard elements. This change does not affect the PNPS Emergency Plan. No further evaluation is required for this change.
2	Revise description of earthquake by removing: ", and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated."	EAL HU1.1, Page 70	No	No	N/A	NO - The NEI 99-01 Revision 5 EAL that provides guidance for the site specific EAL(s) is comprised of three conditions (two of which are required to declare an Unusual Event) and reads as follows: "HU1 EAL#1 "Seismic event identified by ANY 2 of the following: • Seismic event confirmed by (site specific indication or method) • Earthquake felt in plant • National Earthquake Center" This change revises the basis statement attributed to a 1989 EPRI document that describes a "felt earthquake." The basis statement describes a felt earthquake as (a) vibratory ground motion felt at the site and recognized as an earthquake based on a consensus of control room operators on duty and (b) the activation of seismic switches (for plants with operable seismic instrumentation). The construction of the EAL itself (shown above) requires two of three conditions in order to declare the Unusual Event. Two of these three conditions are: 1) seismic switch operation (site specific indication or method - bullet one above) and 2) felt earthquake (bullet 2 above).

Revision Matrix for 50.54(q): Review of Changes

Revision 6 of EAL Technical Bases Document

Item #	Description of Change	Section	Editorial Change per Reg Guide 2.129?	Affect on 10 CFR 50.47(b) Planning Standards or NUREG 0654 program elements ?	Planning standard affected	Basis for Conclusion
2 cont						The basis document information is therefore not intended to modify the EAL to "define" a felt earthquake as BOTH vibratory ground motion felt and recognized as an earthquake AND seismic instrumentation activation. This would take an EAL requiring two of three conditions to make the declaration and render it as two of two conditions to require the declaration should decision makers interpret both the felt condition and seismic instrumentation activation as needed to meet the condition of a felt earthquake. This proposed change is being made to prevent such a misinterpretation of the basis document information and consequently to prevent the potential failure to classify an Unusual Event condition. The proposed change removes the basis document phrase "...and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated" that is part of the EPRI description of a felt earthquake.

Revision Matrix for 50.54(q): Review of Changes

Revision 6 of EAL Technical Bases Document

Item #	Description of Change	Section	Editorial Change per Reg Guide 2.129?	Affect on 10 CFR 50.47(b) Planning Standards or NUREG 0654 program elements?	Planning standard affected	Basis for Conclusion
2 cont						This modification of the basis document is acceptable because it in no way changes the meaning or intent of the EAL, where site specific seismic switch operation (bullet 1 above) is already included as one of the three conditions that factor into the Unusual Event declaration and is clearly defined in the site-specific rendering of the EAL itself. The remainder of the basis document statement that refers to a consensus of control operators to determine whether or not a felt earthquake has occurred is retained and needed because it informs and clarifies the EAL condition of an earthquake felt in plant (bullet 2 above). This proposed change does not reduce the effectiveness of the emergency plan because: 1. The change does not modify the meaning or intent of the EAL 2. The change does not result in any additional emergency classifications or any less emergency classifications than those that should be made using the EAL 3. The change does prevent the potential for a misinterpretation of the basis document information that could lead to the failure to classify an Unusual Event when an earthquake is felt in plant and confirmed by the National Earthquake Center, but seismic switches do not activate.

Revision Matrix for 50.54(q): Review of Changes

Revision 6 of EAL Technical Bases Document

Item #	Description of Change	Section	Editorial Change per Reg Guide 2.129?	Affect on 10 CFR 50.47(b) Planning Standards or NUREG 0654 program elements ?	Planning standard affected	Basis for Conclusion
3	Add: "If core uncovery is threatened, the EOPs specify alternate, more extreme RPV level control measures in order to restore and maintain adequate core cooling, including depressurization and restoration with low level pumps. Determination of "restore and maintain" is based on the actions driven by Emergency Operating Procedures to restore level. The inability to reverse the RPV level lowering trend after lining up injection sources and injecting, including the use of low pressure systems following an Emergency Depressurization, would warrant classification."	Table F-1 Fission Product Barrier Matrix, Item 5, Page 268 and Item 7, Page 282	No	No	N/A	NO - This change does not change the intent of the NEI 99-01 Revision 5 scheme: NEI RCS barrier loss criterion 3.B is applicable when emergency depressurization is required. The loss or potential loss of the RCS or Fuel Clad barrier requires the declaration of an Alert. NEI Fuel Clad barrier potential loss criterion 2.A and RCS barrier loss criterion 2.A are both applicable when vessel level cannot be restored and maintained above the top of active fuel. Because there is a loss or potential loss of two barriers in this instance, a Site Area Emergency declaration is required. The proposed basis document change preserves and reinforces the intent of the NEI scheme by progressing the event from an Alert (for emergency depressurization) to a Site Area Emergency if low pressure systems do not recover vessel level. This change does not change intent, facilities, equipment or processes for this procedure or affect any planning standard elements. This change does not affect the PNPS Emergency Plan. No further evaluation is required for this change.
4	Embolden: "Reference is made to PNPS Security Force because these individuals are the designated personnel onsite 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 PNPS Safeguards Contingency Plan."	HU4.1, Page 115.	Yes	No	N/A	NO - This change is an editorial change where the wording is now emboldened This change does not change intent, facilities, equipment or processes for this procedure or affect any planning standard elements. This change does not affect the PNPS Emergency Plan. No further evaluation is required for this change.

Revision Matrix for 50.54(q): Review of Changes

Revision 6 of EAL Technical Bases Document

Item #	Description of Change	Section	Editorial Change per Reg Guide 2.129?	Affect on 10 CFR 50.47(b) Planning Standards or NUREG 0654 program elements?	Planning standard affected	Basis for Conclusion
5	Embolden: "These individuals are the designated onsite 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 controls placed on the PNPS Safeguards Contingency Plan."	HU4.1, Page 116.	Yes	No	N/A	NO - This change is an editorial change where the wording is now emboldened This change does not change intent, facilities, equipment or processes for this procedure or affect any planning standard elements. This change does not affect the PNPS Emergency Plan. No further evaluation is required for this change.
6	Add: "Temporary lineups to allow heat to be absorbed in the Torus are not considered a long term heat sink."	HG4.1, Page 126	No	No	N/A	NO - This is an administrative change resulting in reinforcement of the existing wording to confirm understanding that the term heat sink refers to the ability to transfer heat to the ocean. This added wording was determined through discussion with Operations to ensure consistent understanding of the existing protocol. This change does not change intent, facilities, equipment or processes for this procedure or affect any planning standard elements. This change does not affect the PNPS Emergency Plan. No further evaluation is required for this change.
7	Add: "The "Judgment" classifications should not be used if an applicable EAL classification has already been determined as included in the EAL chart."	EALs HU6.1, HA6.1, HS6.1 and HG6.1, and Table F-1 Fission Product Barrier Matrix elements 4, 6, 12, 15, 21 and 27 on Pages 131, 133, 135, 137, 278, 280, 299, 301, 324 and 326	No	No	N/A	NO - This is an administrative change resulting in reinforcement of the existing wording to confirm understanding that the application of other EALs should take precedence over the "Judgment" determinations. This added wording was determined through discussion with Operations to ensure consistent understanding of the existing protocol. This change does not change intent, facilities, equipment or processes for this procedure or affect any planning standard elements. This change does not affect the PNPS Emergency Plan. No further evaluation is required for this change.

Revision Matrix for 50.54(q): Review of Changes

Revision 6 of EAL Technical Bases Document

Item #	Description of Change	Section	Editorial Change per Reg Guide 2.129?	Affect on 10 CFR 50.47(b) Planning Standards or NUREG 0654 program elements ?	Planning standard affected	Basis for Conclusion
8	Add: "Fires that are not within an H-1 area do not meet the criteria for an UE classification. If a fire alarm in an H-1 area cannot be disproved within the 15 minutes allowed, the UE needs to be performed."	HU2.1, Page 99	No	No	N/A	NO - This is an administrative change resulting in reinforcement of the existing wording to confirm understanding that the application of this EAL is strictly limited to those which occur within an H-1 area. This added wording was determined through discussion with Operations to ensure consistent understanding of the existing wording. This change does not change intent, facilities, equipment or processes for this procedure or affect any planning standard elements. This change does not affect the PNPS Emergency Plan. No further evaluation is required for this change.
9	Add: "Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source. "	HU1.4, Page 77	No	No	N/A	NO - This is an administrative change that exactly repeats existing wording within the NEI 99-01 Basis for EAL HA1.5, also a flooding event. This repetition confirms and underscores consistent understanding of the applied criteria. This change does not change intent, facilities, equipment or processes for this procedure or affect any planning standard elements. This change does not affect the PNPS Emergency Plan. No further evaluation is required for this change.

Attachment 2

Letter Number 2.16.013

EP-IP-261, Technical Support Center (TSC) Operations, Revision 8 and associated 10 CFR 50.54(q) Review

Technical Support Center (TSC) Operations

TYPE H8.24

Change Statement

- Change "TSC IT Specialists" to "TSC IT Specialist". Page 2
- Add step for the Radiological Coordinator to ensure that the Rad/Chem Coordinator is appropriately controlling ALL entrances to the TSC/OSC facility. Page 35
- Change "OSC/TSC" to "TSC/OSC". Page 35
- Add Emergency Telephone Directory Section 6 title "Communication System Instructions/Diagrams". Page 72
- Add checking business "Monitors and Printers" to Attachment 9.15, TSC/OSC Equipment Operation. Page 73
- Add Step "Open the Business Network Computer Cabinet in the Operations Area. Ensure that the business network computer is available for use." Page 73
- Change "Ensure the business network computers are available." to "Ensure that the other business network computers, monitors and printers are available for use, including the monitor in the OSC Pool Area." Page 73
- Add Section "11.0 Disaster Recovery Computer and Printer" with step "[1] Ensure that the Disaster Recovery Computer and printer in the Operations Area are available for use." Page 74
- Change "ERDS Cabinet" to "Business Network Computer Cabinet". Page 74
- Move old Attachment 9.15 TSC UPS (Uninterruptible Power Supply) System Section 11.0 to 14.0. Page 75
- Delete "EP File 1.6.4, DAMAGE version 2.0 Computer Application Verification, Validation and Documentation 11/22/95" from Document Cross-References. Page 84

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TABLE OF CONTENTS (Cont.)

1.0 PURPOSE

To describe the activation and operation of the PNPS Technical Support Center (TSC).

2.0 REFERENCES

[1] 10CFR50.54(x)

[2] EN-AD-103, *"Document Control and Records Management Programs"*

[3] EN-EP-311, *"Emergency Response Data System (ERDS) Activation via the Virtual Private Network (VPN)"*

[4] EN-EP-601, *"Corporate Emergency Center Operations"*

[5] EN-EP-801, *"Emergency Response Organization"*

[6] EN-LI-102, *"Corrective Action Process"*

[7] EN-NS-102, *"Fitness for Duty Program"*

[8] EN-RP-104, *"Personnel Contamination Events"*

[9] EP-IP-100, *"Emergency Classification and Notification"*

[10] EP-IP-100.1, *"Emergency Action Levels (EALs)"*

[11] EP-IP-261, Attachment 9.21, Document Cross-Reference

[12] EP-IP-400, *"Protective Action Recommendations"*

[13] EP-IP-420, *"Search and Rescue"*

[14] EP-IP-520, *"Transition and Recovery"*

[15] EP-PP-01, "Pilgrim Nuclear Power Station (PNPS) Emergency Plan"

[16] PNPS 2.6.1, "*Emergency and Plant Information Computer (EPIC) System Displays*"

3.0 DEFINITIONS

None

4.0 RESPONSIBILITIES

[1] The first members of the Technical Support Center (TSC) staff to arrive and the entire staff are responsible for setting up the facility using this Procedure.

[2] The Emergency Plant Manager (EPM) has the responsibility for the command and control of all accident mitigation actions at the site, performs these duties from the Technical Support Center (TSC), reports to the Emergency Director, and is responsible for:

(a) Overall management of the onsite Emergency Response Organization (ERO).

(b) Directing the activities of the TSC until the accident is terminated.

(c) Coordinating with the Shift Manager (SM) and the Emergency Director (ED) those response actions necessary for control of the accident, including protection of emergency personnel and the public.

(d) Peer checking emergency classifications.

(e) Ensuring all resources are available to mitigate emergency conditions. Keeping the Emergency Director informed on the status of the plant and conditions within the Protected Area fence.

(f) Authorizing extension of emergency exposures to onsite personnel. (Non-delegable responsibility)

(g) Authorizing the use of potassium iodide (KI) to onsite personnel. (Non-delegable responsibility)

(h) Authorizing all TSC and OSC activities involving the dispatch of personnel from any onsite emergency response facility. (Non-delegable)

(i) Site access authorization for ERO personnel who have been determined to have a BAC of 0.04 or greater. (Non-delegable responsibility)

(j) Waiving initial requirements for access authorization to PNPS. (Non-delegable responsibility)

(k) Coordination and conduct of plant accident assessment and mitigating activities associated with PNPS operations (the Shift Manager retains the responsibility for actual operation of the plant).

(l) The direction of actions taken for the protection for onsite PNPS personnel, including site evacuation activities.

(m) Coordination and conduct of emergency classification recommendations.

(n) Requesting assistance from federal and state agencies, if required.

[3] The TSC Manager is responsible for the performance of technical assessments and communicating the conclusions to the EPM.

[4] The TSC Communicator reports to the TSC Manager and is responsible for maintaining the facility log on WebEOC or other method, ensuring timeliness of facility briefs and, if SPDS is unavailable, updating the Plant Data and Event Chronology Status Boards as information becomes available from the Plant Data Phone (PDP).

[5] The TSC Security Coordinator reports to the EPM and is responsible for the coordination of assembly, accountability, and Station evacuation; emergency access to vital areas and physical security of the Station; overall coordination of the offsite assistance for the security related response; and is the designated liaison between the ICP Security Coordinator at the Incident Command Post (ICP) and site organization.

[6] The Engineering Coordinator reports to the TSC Manager and is responsible for coordinating engineering work requests with the Engineering support team and supporting SAG/SAM decision making and ES/TAG activities.

[7] The Radiological Coordinator reports to the TSC Manager and is responsible for radiological assessments and the development of radiological plans; making recommendations regarding ECL and onsite protective and corrective actions; ensuring that personnel are decontaminated, if necessary; tracking dose received by emergency workers and recommending authorization of personnel exposures below 5 rem TEDE; advising with regard to the use of potassium iodide (KI) onsite; and directing briefing, dispatch, and control of the Personnel Monitoring Team sent from the TSC to the designated Assembly Area for a PA evacuation, if performed.

[8] The Operations Coordinator reports to the TSC Manager and coordinates TSC efforts in determining the nature and extent of emergencies pertaining to equipment and plant facilities in support of Control Room actions, assists the Work Control Coordinator in determining the priority assigned to OSC activities, initiates immediate corrective actions to invoke the provisions of 10CFR50.54(x) if appropriate, and specifically when addressing Severe Accident Management Guidelines (SAMG/SAG) ensures the Control Room, the TSC, and the EOF are informed of significant changes in event status.

[9] Engineers (Mechanical, Electrical, I&C, and Operations) report to the Engineering Coordinator and respond to engineering requests from the Engineering Coordinator and support SAM/SAG as needed.

[10] The Reactor Engineer reports to the Engineering Coordinator and is responsible for supporting the TSC in calculating and tracking core reactivity (core damage assessment), completing core damage calculations, and overall estimation of core damage

[11] The TSC IT Specialist reports to the Engineering Coordinator and is responsible for monitoring the facility equipment (computer-related and communications) to ensure adequate operation, and, if requested by the Control Room, performs verification/activation of the Emergency Response Display System (ERDS) link with the NRC.

5.0 DETAILS

5.1 PRECAUTIONS AND LIMITATIONS

None

5.2 PROCEDURE

NOTE
Steps may be performed concurrently or out of sequence as deemed appropriate.

[1] The Emergency Plant Manager shall follow the instructions outlined in Attachment 9.1, Emergency Plant Manager.

[2] The TSC Manager shall follow the instructions outlined in Attachment 9.2, TSC Manager.

[3] The TSC Communicator shall follow the instructions outlined in Attachment 9.3, TSC Communicator.

[4] The TSC Security Coordinator shall follow the instructions outlined in Attachment 9.4, TSC Security Coordinator.

[5] The Radiological Coordinator shall follow the instructions outlined in Attachment 9.5, Radiological Coordinator.

[6] The Engineering Coordinator shall follow the instructions outlined in Attachment 9.6, Engineering Coordinator.

[7] The Reactor Engineer shall follow the instructions outlined in Attachment 9.7, Reactor Engineer.

[8] The Operations Coordinator shall follow the instructions outlined in Attachment 9.8, Operations Coordinator.

[9] The TSC Engineers shall follow the instructions outlined in Attachment 9.9, TSC Engineers.

[10] The TSC IT Specialist shall follow the instructions outlined in Attachment 9.10, TSC IT Specialist.

[11] To fill an ERO vacancy during facility activation or operation, refer to Attachment 9.20, Filling An ERO Vacancy During Facility Activation/Operation.

6.0 INTERFACES

[1] EMG-100, *"Emergency Management Guideline"*

[2] EN-AD-103, *"Document Control and Records Management Programs"*

[3] EN-RP-104, *"Personnel Contamination Events"*

[4] EP-AD-100, *"Emergency Preparedness Controlled Documents and Record Management Controls"*

[5] EP-IP-225, *"Severe Accident Management Support"*

[6] EP-IP-240, *"Emergency Security Organization Activation and Response"*

[7] EP-IP-315, *"Personnel Monitoring Team Activation and Response"*

[8] EP-IP-330, *"Core Damage"*

[9] EP-IP-410, *"Evacuation/Assembly"*

[10] EP-IP-440, *"Emergency Exposure Controls"*

[11] PNPS 2.2.134, "*TSC - HVAC System Normal and Postaccident Operation*"

[12] PNPS 3.M.3-22, *"TSC Uninterruptible Power Supply System Maintenance"*

[13] PNPS 3.M.3-23, *"TSC Electrical Diesel System Maintenance - Critical Maintenance"*

[14] PNPS Emergency Telephone Directory, Section 6.0.

7.0 RECORDS

Any logs or forms completed by members of the Emergency Response Organization (ERO) during an actual declared emergency are permanent quality records and are maintained in accordance with EP-AD-100, *"Emergency Preparedness Controlled Documents and Record Management Controls"*, and EN-AD-103, *"Document Control and Records Management Programs"*.

8.0 REQUIREMENTS AND COMMITMENTS

This section lists those external commitments (i.e., NRC commitments, QA audit findings, and INPO inspection items) implemented in this Procedure.

Reference Document	Commitment	Affected Sections/Steps and Former Procedures	Standard Procedure
NRC Inspection Finding 86-39-01	Maintain the chronology status board in the TSC up-to-date.	EP-IP-220 TSC Activation	EP-IP-261 TSC Operations Attachment 9.2 TSC Manager Attachment 9.3 TSC Communicator
NRC Inspection Finding 81-15-47	Develop procedures for use during emergencies which describe the concept of operations of the emergency repair and corrective action teams, including reporting chains and precautions appropriate for the situation.	EP-IP-230 OSC Activation and Response All	EP-IP-262 OSC Operations Attachment 9.4 Work Control Coordinator

9.0 ATTACHMENTS

9.1 EMERGENCY PLANT MANAGER
9.2 TSC MANAGER
9.3 TSC COMMUNICATOR
9.4 TSC SECURITY COORDINATOR
9.5 RADIOLOGICAL COORDINATOR
9.6 ENGINEERING COORDINATOR
9.7 REACTOR ENGINEER
9.8 OPERATIONS COORDINATOR
9.9 TSC ENGINEERS
9.10 TSC IT SPECIALIST

9.11 EMERGENCY TASK ASSIGNMENT SHEET AND DIRECTIONS
9.12 EMERGENCY TASK TRACKING FORM
9.13 AREA RADIATION MONITOR DATA FORM
9.14 PROCESS RADIATION MONITOR DATA FORM
9.15 TSC/OSC EQUIPMENT OPERATION
9.16 TSC PRESSURE BOUNDARY/LOCATION MAP
9.17 PLANT DATA FORM
9.18 TSC BRIEFING GUIDELINE - PERIODIC BRIEF
9.19 ENGINEERING ACTION ITEM TRACKING FORM
9.20 FILLING AN ERO VACANCY DURING FACILITY ACTIVATION/OPERATION
9.21 DOCUMENT CROSS-REFERENCE

ATTACHMENT 9.1 **EMERGENCY PLANT MANAGER**

Sheet 1 of 8

Name: ______________________________ Date: ___________
Emergency Plant Manager

NOTE

Steps may be performed concurrently or out of sequence as deemed appropriate.

1.1 Actions For Initial Response

1.2 Facility Activation

Notes

NOTE

A goal of 60 minutes for the facility to be operational under optimum conditions has been established for the TSC.

[1] Report as directed by the ERO notification message.

(a) **IF** responding to the site, **AND** depending on the nature of the emergency, **THEN** determine whether to go the Control Room to speak directly with the on-shift Emergency Director or to report to the TSC and contact the Control Room by phone.

(1) Use an Essential Information Checklist (obtain from the Operations Coordinator if available) and review each of the listed areas, including any teams dispatched from the Control Room. Verify that the appropriate classification was made.

(2) Check the status of any protective action recommendations (applicable for a General Emergency only).

ATTACHMENT 9.1 **EMERGENCY PLANT MANAGER**

Sheet 2 of 8

Notes

(b) **IF** responding to a security event **AND** directed to the Chiltonville Training Center (CTC) as a staging area, **THEN** perform the following:

(1) Report to the CTC and ensure all reporting individuals report inside the building.

(2) Contact the Control Room by phone. Use an Essential Information Checklist to review each of the listed areas and verify the classification declared, any protective action recommendations (for a General Emergency only), and any teams dispatched from the Control Room.

(3) Establish command and control using the simulator and available documents, Procedures, drawings, etc., as needed.

(4) Determine supplies and phones available for use and provide contact information to the EOF Manager, Control Room, and Incident Command Post (ICP).

(5) Assign specific individuals to perform TSC/OSC positions and support functions within the staging area location.

[2] Upon arrival at the TSC or alternate location, sign in on the TSC roster board and/or sign-in sheet. Ensure the TSC Communicator and TSC Manager perform the following:

(a) TSC Communicator: Maintain a log of all pertinent actions and decisions made during the course of the response. Update Plant Data Form as necessary.

(b) TSC Manager: Complete an Emergency Response Organization Roster for the TSC, Security, and the Control Room.

Notes

[3] Instruct the TSC Manager to ensure clocks are synchronized with the official Control Room time.

[4] **WHEN** you have been sufficiently briefed and assured that an adequate staff has assembled as indicated by the TSC Manager, the OSC Manager, and the Radiological Coordinator, **THEN**:

(a) Inform the Emergency Director (either in the Control Room or the EOF depending on operational status) you have assumed the Emergency Plant Manager position.

(b) Inform the Shift Manager that:

(1) The TSC and OSC are operational and prepared to assume responsibility for support activities.

(2) All shift personnel not directly involved in control of the plant from within the Control Room shall be sent to the OSC Operations Support.

(c) For the initial brief to the TSC/OSC staff, use the Essential Information Checklist.

(d) For periodic briefs, use Attachment 9.18,TSC Briefing Guideline - Periodic Brief.

2.1 Facility Operation

Notes

2.2 Operation

> NOTE
>
> EMG-100, *"Emergency Management Guideline"*, may be used to assist in assessment of plant accident conditions and mitigating activities.

[1] Discuss the situation with the Emergency Director and the EOF Manager when they arrive at the EOF.

[2] Authorize extending emergency radiation exposure and the use of potassium iodide (KI) for onsite personnel. Refer to EP-IP-440, *"Emergency Exposure Controls"*, for information and guidance on emergency exposure controls.

[3] Waive initial requirements for access authorization as needed.

[4] Periodically confer with the Emergency Director to review the status of the situation and progress toward resolution.

[5] Periodically confer with the following individuals to review actions being implemented, priorities of assigned activities, and to ensure a coordinated response by each group:

(a) TSC Manager

(b) Radiological Coordinator

(c) OSC Manager

(d) Emergency Plant Operations Supervisor

(e) TSC Security Coordinator

Notes

[6] Ensure that periodic updates are provided to the onsite organization, as appropriate, using the facility public-address (PA) system.

[7] Ensure documentation for assigned activities is maintained and completed by TSC and OSC personnel.

NOTES

1. Nonessential contractor personnel, handicapped personnel, and visitors are sent offsite at an Alert. All work not in direct support of mitigating the emergency should be suspended and workers ordered to return to their staging areas (shops, trailers, offices, etc.).
2. Accountability as well as Protected Area evacuation for all nonessential personnel are required at a Site Area Emergency or General Emergency unless restricted by Security or other events affecting safe movement of persons from the site.

[8] Determine whether site access authorization will be given for ERO personnel who have been identified to have a BAC of 0.04 or greater, specifically with regard to:

(a) The individual's condition.

(b) The needs of the response effort in support of the emergency.

(c) The determination of when and in what capacity the individual shall be used.

Notes

[9] At a Site Area Emergency or a General Emergency (or whenever deemed necessary):

(a) Order a site evacuation and direct Security to prohibit access of nonessential personnel to the site. Evacuation and assembly are conducted in accordance with EP-IP-410, "*Evacuation/Assembly*".

(b) Ensure personnel accountability is performed and maintained inside the Protected Area. **IF** personnel are determined to be missing, **THEN** initiate search and rescue in accordance with EP-IP-420, "*Search and Rescue*".

[10] **IF** a localized emergency exists (that is, one that affects only a portion of a building or the site), **THEN** direct Security to control access to the area(s).

[11] Continuously monitor plant conditions and ensure that the Emergency Director remains apprised of specific circumstances which impact the emergency classification and protective measures. Refer to EP-IP-100, "*Emergency Classification and Notification*", for information and guidance on emergency classification.

[12] Refer to Attachment 9.11, Emergency Task Assignment Sheet and Directions, for information and guidance on task assignment and support activities.

[13] **IF** required, **THEN** request assistance from federal and state agencies through the EOF Manager.

Notes

2.3 TSC/OSC Relocation

[1] **IF** it becomes necessary to evacuate or relocate staff from the TSC/OSC, **THEN** take the following actions:

(a) Direct the OSC Manager to coordinate the relocation of the OSC staff to the Control Room Annex.

(b) Identify the minimum TSC staff needed for the conditions at hand and relocate with this staff to the Control Room.

(c) Direct the TSC Manager to coordinate the relocation of the remaining TSC staff to the Emergency Operations Facility (EOF) in accordance with steps in Attachment 9.2, TSC Manager.

Notes

2.4 Position Turnover

[1] **IF** you are being relieved at the end of your shift or your position is otherwise being turned over, **THEN** perform a formal turnover as follows:

(a) Review status boards, WebEOC facility log, and other information with the position replacement as appropriate.

(b) Provide a briefing on the emergency, radiological conditions, and any actions that have been completed or are in progress.

(c) Inform the position's immediate supervisory position and affected staff that you are transferring responsibility for the position to the replacement individual.

(d) Sign off at the facility roster board, ensure your replacement signs in, and turn over the position binder and facility badge to the new position replacement.

3.1 Facility Deactivation

3.2 Termination

[1] Direct the TSC Manager to proceed with facility deactivation according to Attachment 9.2, TSC Manager.

[2] Return all equipment to its proper storage locations.

[3] Review all documentation maintained during the emergency.

[4] Provide all logs, forms, and records to the Emergency Planning Manager upon termination of the emergency and entry into the Recovery Phase.

ATTACHMENT 9.2 **TSC MANAGER**

Sheet 1 of 7

Name: ______________________________ Date__________

TSC Manager

NOTE
Steps may be performed concurrently or out of sequence as deemed appropriate.

1.1 Actions For Initial Response

Notes

1.2 Facility Activation

[1] Report to the TSC and sign in on the TSC roster board.

[2] Begin and maintain a log of your work activities.

[3] Assume command of the TSC.

NOTE
A goal of 60 minutes for the facility to be operational under optimum conditions has been established for the TSC.

[4] Upon their arrival, direct the Engineering Coordinator and Operations Coordinator to activate the TSC.

(a) **IF** WebEOC is unavailable, **THEN** ensure the TSC Communicator is updating the Plant Data Form and Event Chronology Status Boards **[NRC Inspection Finding 86-39-01]** as information becomes available. The Plant Data Form and facility log boards can be maintained via WebEOC, if available.

Notes

(b) Inform the Emergency Plant Manager that the TSC is ready to be declared operational when the following:

(1) Minimum TSC staff positions have been filled. The minimum staffing for operation of the TSC includes the following. To fill a vacancy, refer to Attachment 9.20, Filling An ERO Vacancy During Facility Activation/Operation.

a. Emergency Plant Manager

b. TSC Manager

c. Engineering Coordinator

d. Operations Coordinator

e. Radiological Coordinator

(2) Information is available to obtain and assess plant status conditions. (This includes having at least one of the following: SPDS, PDP, Mitigation Line, or other available means.)

(3) The TSC ventilation system is in appropriate operational mode. This action will be assigned to Operations Support personnel in the OSC.

ATTACHMENT 9.2 **TSC MANAGER**

Sheet 3 of 7

Notes

NOTE

Prioritization of staffing positions should be based on the nature of the emergency and current and projected emergency response work activities.

2.1 Facility Operation

2.2 Operation

[1] In addition to the minimum personnel necessary to be operational, obtain status of the Engineers from the Engineering Coordinator and check roster sign-in board to ensure all assigned positions are filled. To fill a vacancy, refer to Attachment 9.20, Filling An ERO Vacancy During Facility Activation/Operation.

[2] Through discussions with the Emergency Plant Manager, obtain information on Station status and the tasks expected to be developed given the current situation.

[3] Direct technical and operational assessment activities of the TSC staff in accordance with Attachment 9.11, Emergency Task Assignment Sheet and Directions, as required to support the Control Room operating staff and ERO personnel in responding to the event.

[4] Upon entry into EOPs, ensure that EP-IP-225, "*Severe Accident Management Support*", is being implemented by the Operations Coordinator in the TSC to support EOP implementation and accident mitigation strategies as needed.

Notes

[5] Provide the Emergency Plant Manager with:

(a) Recommendations for changes in emergency classification based on assessment of current and projected plant status in relation to the Emergency Action Levels (refer to EP-IP-100, *"Emergency Classification and Notification"*).

(b) TSC task status updates and priorities.

(c) Information on equipment operational problems and alterations in plant systems operations or lineups.

(d) At SAE, General Emergency, or when deemed necessary, discuss the need for a site evacuation with the EPM in accordance with EP-IP-410, "*Evacuation/Assembly*."

[6] Provide periodic updates on plant status and anticipated actions to all TSC personnel.

[7] Ensure that the TSC Communicator provides reminders to ensure the timeliness of facility briefs, maintains status boards up-to-date, and ensures that current plant data is being made available.

[8] Should there be any indication of actual or potential fuel damage, direct the Engineering Coordinator to ensure core damage assessment is being done in accordance with EP-IP-330, "*Core Damage*".

Notes

[9] **IF** additional TSC personnel are necessary, **THEN** contact the Administration and Logistics Coordinator in the EOF and identify the individuals and/or organizations which are required.

(a) Specific disciplines to be considered include, but are not limited to, Fire Protection, RP, Materials and Warehouse personnel.

[10] At a Site Area Emergency or General Emergency, ensure accountability is being conducted by Security. Accountability is to be completed within 30 minutes of the Site Area Emergency or General Emergency.

2.3 Relocation

[1] **IF** it becomes necessary to evacuate or relocate staff from the TSC, as determined by the Emergency Plant Manager (EPM), **THEN** take the following actions:

(a) Using available staff, support the relocation of the EPM, the Operations Coordinator, Operations Support, the TSC Security Coordinator, and any additional personnel identified by the EPM to the Control Room.

(b) Contact the Administration and Logistics Coordinator at the EOF and request that space be made available for the remaining TSC staff.

(c) Direct the TSC staff to gather any logs and records needed to continue emergency operations from the EOF and to begin relocation to the EOF.

Notes

[2] When relocation to the EOF is completed, establish communications with the EPM in the Control Room and continue to coordinate TSC Operations.

[3] As deemed appropriate, the TSC staff will gather necessary support material from the Support Building to be made available at the EOF for the TSC staff.

2.4 Position Turnover

[1] **IF** you are being relieved at the end of your shift or your position is otherwise being turned over, **THEN** perform a formal turnover as follows:

(a) Review status boards, WebEOC facility log, and other information with the position replacement as appropriate.

(b) Provide a briefing on the emergency, radiological conditions, and any actions that have been completed or are in progress.

(c) Inform the position's immediate supervisory position and affected staff that you are transferring responsibility for the position to the replacement individual.

(d) Sign off at the facility roster board, ensure your replacement signs in, and turn over the position binder and facility badge to the new position replacement.

ATTACHMENT 9.2 **TSC MANAGER**

Sheet 7 of 7

3.1 Facility Deactivation

Notes

3.2 Termination

[1] Upon direction from the Emergency Plant Manager, deactivate the TSC by:

(a) Terminating assessment activities and communications via communication lines.

(b) Deactivating all TSC equipment and placing it in its pre-emergency condition.

(c) Ensuring the TSC ventilation system is secured by the Operations Coordinator in accordance with Attachment 9.15, TSC/OSC Equipment Operation.

(d) Returning the PNPS controlled Procedures to the TSC/OSC bookcase storage area.

(e) Collecting and forwarding all logs and records to the Emergency Plant Manager.

(f) Closing out the TSC log by noting the time that the TSC was deactivated.

[2] The TSC Manager shall report any equipment, Procedure, or personnel problems to the Emergency Plant Manager.

ATTACHMENT 9.3 **TSC COMMUNICATOR**

Sheet 1 of 2

Name: ______________________________ Date:__________

TSC Communicator

NOTE
Steps may be performed concurrently or out of sequence as deemed appropriate.

1.1 Actions For Initial Response

Notes

1.2 Facility Activation

[1] Report to the TSC and sign in on the TSC roster board.

[2] Begin and maintain a log of major events, decisions, communications, and actions at the TSC using the facility log via WebEOC, if available, or manually.

[3] Report to the TSC Manager and obtain the Emergency Classification and plant status.

2.1 Facility Operation

2.2 Operation

[1] Maintain the facility log via WebEOC of all pertinent actions and decisions.

[2] Obtain plant data needed via SPDS as possible.

[3] **IF** WebEOC is unavailable, **THEN** commence updating the Plant Data and Event Chronology Status Boards **[NRC Inspection Finding 86-39-01]** as information becomes available from the Plant Data Phone (PDP). The Plant Data and facility log boards can be maintained via WebEOC, if available, or Plant Data Form, Attachment 9.17, if WebEOC is not available.

[4] Provide reminders to ensure the timeliness of facility briefs, maintain status and classification boards up-to-date, and ensure that current plant data is being made available.

ATTACHMENT 9.3 **TSC COMMUNICATOR**

Sheet 2 of 2

Notes

2.3 Position Turnover

[1] **IF** you are being relieved at the end of your shift or your position is otherwise being turned over, **THEN** perform a formal turnover as follows:

(a) Review status boards, WebEOC facility log, and other information with the position replacement as appropriate.

(b) Provide a briefing on the emergency, radiological conditions, and any actions that have been completed or are in progress.

(c) Inform the position's immediate supervisory position and affected staff that you are transferring responsibility for the position to the replacement individual.

(d) Sign off at the facility roster board, ensure your replacement signs in, and turn over the position binder and facility badge to the new position replacement.

3.1 Facility Deactivation

3.2 Termination

[1] Return the work area to its pre-emergency condition.

[2] Gather all logs, records, and documents and turn them over to the TSC Manager.

[3] Notify the TSC Manager when your position is deactivated.

ATTACHMENT 9.4 **TSC SECURITY COORDINATOR**

Sheet 1 of 4

Name: ______________________________ Date:__________

TSC Security Coordinator

NOTE
Steps may be performed concurrently or out of sequence as deemed appropriate.

1.1 Actions For Initial Response

1.2 Facility Activation

	Notes
[1] Report to the TSC and sign in on the TSC roster board.	
[2] Begin and maintain a log of your work activities.	
[3] Report to the Emergency Plant Manager and obtain the Emergency Classification and plant status.	
[4] **IF** the event is hostile action based, **THEN** receive and maintain communication with the ICP Security Coordinator to establish communications and level of activity.	

2.1 Facility Operation

Notes

2.2 Operation

[1] Through discussions with the Emergency Plant Manager, obtain information on current situation.

CAUTION

IF an emergency classification is entered due to a security condition, **THEN** evacuation and accountability may put personnel at risk. Therefore, in these situations, evacuation and accountability will be suspended until directed by Security and cleared through the Incident Commander.

[2] **IF** the event is hostile action based, **THEN** coordinate all site response actions with the ICP Security Coordinator.

[3] Direct activities in accordance with EP-IP-240, "*Emergency Security Organization Activation and Response*", as required in responding to the event, including:

(a) Managing the Emergency Security Organization.

(b) Keeping the Emergency Plant Manager informed on security issues including any fitness for duty concerns of responding emergency response personnel.

(c) Providing Security assistance during facility activation, accountability, evacuation and assembly (EP-IP-410, *"Evacuation/Assembly"*), and search and rescue (EP-IP-420, *"Search and Rescue"*).

(d) Scheduling long-term Security operations.

ATTACHMENT 9.4 **TSC SECURITY COORDINATOR**

Sheet 3 of 4

Notes

(e) Coordinating offsite Security activities with the MEMA Region II EOC and the Plymouth Police Department through the ICP Security Coordinator, if appropriate.

(f) Providing Security support to the Administration and Logistics Coordinator, including the EOF, if requested.

[4] During a Site Area Emergency or General Emergency it may become necessary to evacuate the Main Gate/BREs due to habitability concerns.

(a) **IF** notified by the EPM, **AND** as confirmed by the Radiological Coordinator of the need to evacuate the Main Gate/BREs, **THEN** the Security Operations Supervisor will:

(1) Advise the ICP Security Coordinator of the pending actions.

(2) Assess relocation alternatives with the EPM.

(3) Develop a plan for the relocation of Security personnel and equipment

(4) Determine necessary compensatory measures and actions as a result of the relocation.

(5) Engage evacuation activities as needed.

Notes

2.3 Position Turnover

[1] **IF** you are being relieved at the end of your shift or your position is otherwise being turned over, **THEN** perform a formal turnover as follows:

(a) Review status boards, WebEOC facility log, and other information with the position replacement as appropriate.

(b) Provide a briefing on the emergency, radiological conditions, and any actions that have been completed or are in progress.

(c) Inform the position's immediate supervisory position and affected staff that you are transferring responsibility for the position to the replacement individual.

(d) Sign off at the facility roster board, ensure your replacement signs in, and turn over the position binder and facility badge to the new position replacement.

3.1 Facility Deactivation

3.2 Termination

[1] Upon notification of termination, ensure notification is made to the ICP Security Coordinator.

[2] Return the work area to its pre-emergency condition.

[3] Gather all logs, records, and documents and turn them over to the TSC Manager.

[4] Notify the Emergency Plant Manager when your position is deactivated.

ATTACHMENT 9.5 **RADIOLOGICAL COORDINATOR**

Sheet 1 of 12

Name: ______________________________ Date:__________

Radiological Coordinator

NOTE
Steps may be performed concurrently or out of sequence as deemed appropriate.

1.1 Actions For Initial Response

Notes

1.2 Facility Activation

[1] The on-shift Radiological Supervisor (or Technician if a Supervisor is not available) will implement the following actions:

(a) Follow the directions provided by the ERO notification message or other information that may be received through plant announcements, backup notifications, or manual call-outs.

(b) Contact the Shift Manager (Emergency Director) in the Control Room for a briefing of the emergency situation and any RP actions needed to support initial emergency response activities underway.

(c) **IF** it is determined that TSC/OSC staff activation is necessary **OR** the decision has been made with the initial response notification, **THEN** implement the following actions:

(1) Coordinate initial response activities with the Shift Manager (Emergency Director) in the Control Room or the TSC Manager if present.

(2) Assume the responsibility for onsite radiological controls until relieved by the Radiological Coordinator.

Notes

[2] The Radiological Coordinator will implement the following actions:

(a) Follow the directions provided by the ERO notification message or other information that may be received through plant announcements, backup notifications, or manual call-outs.

(b) **IF** it is determined that TSC/OSC staff activation is necessary **OR** the decision has been made with the initial response notification, **THEN** implement the following actions:

(1) Report to the TSC or designated location, sign in on the TSC roster board, and start maintaining a log of your work activities.

(2) Contact the on-shift RP Supervisor/Technician for a briefing of the emergency situation and activities underway.

[3] Evaluate the need to call out for Radiation Protection support.

[4] Establish a routine review and assessment of area radiation and effluent monitors (via status boards, SPDS, and EPIC).

[5] Interface with other engineers in the TSC concerning present and projected radiological conditions.

ATTACHMENT 9.5 **RADIOLOGICAL COORDINATOR**

Sheet 3 of 12

Notes

[6] Ensure the Rad/Chem Coordinator is performing the following:

(a) Appropriately controlling ALL entrances to the TSC/OSC facility.

(b) Ensuring habitability is maintained within the TSC/OSC facility and other occupied areas within the plant by conducting radiological surveys as conditions warrant.

(c) Obtaining keys from the TSC/OSC key locker and opening the OSC radiological equipment supply lockers. Checking all instruments for operability and performing an inventory of any locker which had a broken seal.

[7] Relieve the on-shift RP Supervisor/Technician of the responsibility for onsite radiological controls.

2.1 Facility Operation

Notes

2.2 Operation

[1] Assign a member of the Radiation Protection staff to ensure all personnel responding to the TSC, OSC, and Control Room without dosimetry are issued a TLD.

NOTE

Sources of meteorological data for the Process Radiation Monitor Data Form include:

1.	220' Met Tower	Control Room or SPDS Graphics Menu (414 PNPS EP Weather Display)
2.	160' Met Tower	Local Indication or EOF MeDAP Computer
3.	Air and marine weather radio forecasts	EOF Marine Radio
4.	National Weather Service (Taunton)	Telephone
5.	The Weather Channel	EOF Cable TV
6.	Visual estimation	Affected Area

[2] Assign one Radiation Data Communicator to the Control Room and one to the TSC to establish communications and track effluent monitor and radiation data utilizing the Area Radiation Monitor Data Form (Attachment 9.13), the Process Radiation Monitor Data Form (Attachment 9.14), and the Emergency Conferencing System Rad Data Phone (see Emergency Telephone Directory Section 6.0).

Notes

[3] Determine the recent exposure history and respirator qualifications of ERO personnel (through DDB, historical files, or by estimation) via Dosimetry Clerk, if available, and distribute as follows:

(a) Provide the Rad/Chem Coordinator with an exposure listing of all OSC supervisory and pool personnel expected to be dispatched.

(b) Provide the Offsite Monitoring Team Coordinator in the EOF with an exposure listing of all personnel involved in offsite plume tracking and environmental monitoring.

[4] Ensure that an individual exposure record is started for all personnel expected to receive exposure over the course of the emergency. Emergency exposure can be tracked on a PNPS Emergency Dose Card (EP-IP-440, *"Emergency Exposure Controls"*) or other similar record.

[5] Brief the TSC Manager on present radiological conditions.

[6] Initially verify accountability of the Radiation Protection staff in the TSC by ensuring that all personnel have signed in on the TSC roster board (and thereafter maintain continuous accountability).

[7] Report to the TSC Manager when the Radiation Protection staff is capable of performing radiological controls activities. The following positions should be staffed in support of emergency activities and functions:

(a) Radiation Protection Technician (1)

(b) Radiation Data Communicator (2)

Notes

2.3 On-Site Radiological Operations

[1] Control onsite personnel exposure, accumulated dose, and the distribution of potassium iodide (KI) during the emergency (see EP-IP-440, "*Emergency Exposure Controls*," for specific guidance).

[2] Direct the coordination of the tracking of emergency radiation exposure to ERO personnel dispatched from the OSC. If personnel are cycled between onsite and offsite activities during the course of the emergency, ensure the Rad/Chem Coordinator provides, and is informed of, personnel accumulated emergency exposures.

(a) Ensure that personnel are decontaminated if necessary.

[3] Assess the status of current and projected onsite radiological conditions and based upon the circumstances:

(a) Identify areas requiring radiological controls and areas containing potential radiological hazards.

(b) Consider issuing SIDs to facility personnel or placing several SIDs throughout occupied areas of the facilities if radiological conditions warrant.

(c) Consider whether to dispatch a Personnel Monitoring Team to prepare the Assembly Area prior to a Protected Area evacuation or when a Site Area Emergency is declared (see Section 2.3 of this Attachment for specific guidance).

Notes

NOTE

The Emergency Plant Manager is to be consulted with concerning any onsite evacuation recommendations.

(d) Determine whether a Protected Area evacuation is necessary and determine the appropriate Assembly Area in accordance with EP-IP-410, *"Evacuation/Assembly"* (automatically determined and initiated at a Site Area Emergency).

(e) Plan for possible selective onsite building evacuations in the event of a release.

[4] Frequently brief the TSC Manager, the Operations Coordinator, and the Radiation Protection staff regarding:

(a) Present and projected onsite radiological conditions.

(b) Recommendations for corrective actions based upon radiological conditions.

(c) Radiological concerns for OSC teams.

[5] In coordination with the OSC Manager, the TSC Manager, and the Radiological Assessment Coordinator, assess staffing of Radiation Protection personnel in the TSC and the OSC. **IF** additional personnel are necessary, **THEN** consider the following:

(a) **IF** personnel are standing by onsite (such as in an Alert), **THEN** obtain support from normal muster, shop, or office locations.

(b) **IF** personnel have been evacuated to an Assembly Area, **THEN** coordinate with the Administration and Logistics Coordinator in the EOF to obtain additional support.

ATTACHMENT 9.5 **RADIOLOGICAL COORDINATOR**

Sheet 8 of 12

Notes

2.4 Dispatch Of The Personnel Monitoring Team

[1] Personnel monitoring is not necessary unless there is a potential for contamination on evacuees.

(a) **IF** evacuees will be exiting contaminated areas without removal of protective clothing, **THEN** dispatch a team to perform monitoring at the Assembly Area.

(b) **IF** emergency conditions do NOT warrant monitoring **AND** evacuees are NOT rapidly exiting any contaminated areas, **THEN** Radiation Protection personnel may be retained for other support purposes.

[2] Generate an Emergency Task Assignment Sheet and Directions form (in accordance with Attachment 9.11) to begin the process of assembling and dispatching the Personnel Monitoring Team. Team members can be taken from:

(a) Radiation Protection OSC pool personnel.

(b) Available Radiation Protection personnel assigned to the EOF.

(c) Evacuated Radiation Protection personnel already at the Assembly Area.

[3] Responsibility and control of the Personnel Monitoring Team may be transferred to the Radiological Assessment Coordinator if desired.

Notes

NOTE

Team briefings may be directed to be conducted by the Radiological Assessment Coordinator or the Assembly Area Coordinator as appropriate.

[4] Ensure the Personnel Monitoring Team briefing will include:

(a) An estimation of the number of individuals to be monitored (if known).

(b) Contamination levels below which personnel and vehicles can be unconditionally released.

(c) Disposition of contaminated individuals and items.

(d) Overall Procedure responsibilities and instructions in EP-IP-315, *"Personnel Monitoring Team Activation and Response"*.

[5] Notify the Administration and Logistics Coordinator in the EOF of the dispatch of the Personnel Monitoring Team to the Assembly Area.

2.5 Relocation Of The TSC/OSC

[1] **IF** habitability survey results require the evacuation of the TSC/OSC, **THEN**:

(a) Notify the TSC Manager that evacuation of the TSC/OSC is warranted based upon habitability survey results.

Notes

(b) Upon direction from the Emergency Plant Manager: relocate the OSC to the Control Room Annex; the EPM, the Operations Coordinator, Operations Support, the TSC Security Coordinator, and any additional personnel identified by the EPM to the Control Room; and remaining TSC staff to the EOF.

(c) <u>IF</u> the backup OSC does NOT meet habitability requirements, **<u>THEN</u>** an alternate location must be selected with respect to:

(1) Adequate access to the plant.

(2) Communications capabilities with other emergency facilities.

(3) Space for at least 25 people.

(4) Access to appropriate reference materials, tools, and equipment.

(d) Coordinate with the TSC Security Coordinator to determine the evacuation route and any recommended protective measures prior to directing the evacuation.

(e) In coordination with the OSC Manager, designate select personnel, necessary emergency equipment, supplies, and documentation to be relocated to the backup or alternate OSC.

[2] Upon establishing the backup or alternate OSC, notify the TSC Manager and await further instructions.

Notes

2.6 Position Turnover

[1] **IF** you are being relieved at the end of your shift or your position is otherwise being turned over, **THEN** perform a formal turnover as follows:

(a) Review status boards, WebEOC facility log, and other information with the position replacement as appropriate.

(b) Provide a briefing on the emergency, radiological conditions, and any actions that have been completed or are in progress.

(c) Inform the position's immediate supervisory position and affected staff that you are transferring responsibility for the position to the replacement individual.

(d) Sign off at the facility roster board, ensure your replacement signs in, and turn over the position binder and facility badge to the new position replacement.

3.1 Facility Deactivation

Notes

3.2 Termination

[1] Ensure that all radiological logs and records are submitted to the TSC Manager.

[2] Debrief the Reactor Engineer and the Rad/Chem Coordinator regarding:

(a) Communication problems.

(b) Adequacy of personnel training or briefings.

(c) Equipment malfunctions or deficiencies.

(d) Adequacy of facilities and materials.

[3] Brief relief personnel regarding any special survey or posting requirements which may still be in effect as a result of the emergency condition.

[4] Report any equipment, Procedure, or personnel problems to the TSC Manager.

ATTACHMENT 9.6 **ENGINEERING COORDINATOR**

Sheet 1 of 4

Name: ______________________________ Date:__________

Engineering Coordinator

NOTE
Steps may be performed concurrently or out of sequence as deemed appropriate.

1.1 Actions For Initial Response

Notes

1.2 Facility Activation

[1] Report to the TSC and sign in on the TSC roster board.

[2] Begin and maintain a log of your work activities and Engineering tasks via WebEOC, if available, or manually.

[3] Obtain briefing from the TSC Manager to coordinate Systems Engineering initial activation and assessment activities.

[4] Coordinate with the TSC Manager to determine, based upon the nature of the event, what additional technical assessment staff should be called out (i.e., which discipline(s) of engineering assistance will be necessary).

[5] Support SAG/SAM decision making.

[6] If requested by the Control Room, direct the TSC IT Specialist to perform a verification/activation of the Emergency Response Display System (ERDS) link with the NRC in accordance with Attachment 9.15, TSC/OSC Equipment Operation.

Notes

NOTE

Prioritization of staffing positions should be based on the nature of the emergency and current and projected emergency response work activities.

[7] Ensure that the Engineering staff is being manned with the following positions:

(a) Reactor Engineer (1) - "staff augmentation" position (30 min)

(b) Mechanical Engineer (1) - "staff augmentation" position (60 min)

(c) Electrical Engineer (1) - "staff augmentation" position (60 min)

(d) I&C Engineer (1)

(e) TSC IT Specialist (1)

2.1 Facility Operation

2.2 Operation

[1] Initially verify accountability of the Engineering staff in the TSC by ensuring that all personnel have signed in on the TSC sign-in board (and thereafter maintain continuous accountability).

[2] Direct plant system-related engineering assessment tasks in accordance with Attachment 9.11, Emergency Task Assignment Sheet and Directions, as requested.

(a) Assign and coordinate TSC engineering tasks using the Engineering Action Item Tracking Form via WebEOC, or if WebEOC is not available, using Attachment 9.19, Engineering Action Item Tracking Form.

Notes

[3] When directed by the TSC Manager, implement EP-IP-330, *"Core Damage"*, and report results to the TSC Manager.

[4] Coordinate with the Administration and Logistics Coordinator in the EOF to coordinate internal engineering assistance or to contact any outside organizations (GE, Bechtel, etc.) from which engineering assistance is required.

(a) Coordinate with the Administration and Logistics Coordinator to make arrangements for:

(1) Protracted technical assessment support (e.g., 24-hour staffing).

(2) Supplies, lodging, and transportation or other support for response personnel.

[5] Periodically review Engineering support efforts with the TSC Manager.

[6] Ensure the TSC IT Specialist:

(a) Monitors facility equipment (computer-related and communications) to ensure adequate operation.

(b) Resolves any IT related malfunctions.

(c) Assists with issues related to WebEOC.

[7] Report any need for additional Systems Engineering personnel to the TSC Manager.

[8] When appropriate, direct the technical assessment staff to develop an outline of engineering actions which will be required for recovery from the event.

ATTACHMENT 9.6 **ENGINEERING COORDINATOR**

Sheet 4 of 4

Notes

[9] Ensure the development and submittal of the Core Damage Summary Report in accordance with EP-IP-330, "*Core Damage*", and in coordination with the Reactor Engineer.

2.3 Position Turnover

[1] **IF** you are being relieved at the end of your shift or your position is otherwise being turned over, **THEN** perform a formal turnover as follows:

(a) Review status boards, WebEOC facility log, and other information with the position replacement as appropriate.

(b) Provide a briefing on the emergency, radiological conditions, and any actions that have been completed or are in progress.

(c) Inform the position's immediate supervisory position and affected staff that you are transferring responsibility for the position to the replacement individual.

(d) Sign off at the facility roster board, ensure your replacement signs in, and turn over the position binder and facility badge to the new position replacement.

3.1 Facility Deactivation

3.2 Termination

[1] Return the work area to its pre-emergency condition.

[2] Gather all logs, records, and documents and turn them over to the TSC Manager.

[3] Notify the TSC Manager when your position is deactivated.

ATTACHMENT 9.7 REACTOR ENGINEER

Sheet 1 of 2

Name: ______________________________ Date:__________

Reactor Engineer

NOTE

Steps may be performed concurrently or out of sequence as deemed appropriate.

1.1 Actions For Initial Response

Notes

1.2 Facility Activation

[1] Report to the TSC and sign in on the TSC roster board.

[2] Begin and maintain a log of your work activities.

[3] Ascertain plant status via Emergency Conferencing System Mitigation Line (see ETD Section 6.0) or SPDS, if available, and start the Damage Program on the Core Damage Computer.

2.1 Facility Operation

2.2 Operation

[1] Through discussions with the Engineering Coordinator and in accordance with EP-IP-330, "*Core Damage*", determine the following:

(a) Obtain reactor power historical information.

(b) Complete core damage calculations and overall estimation of core damage.

(c) Provide core parameter information results back to the Engineering Coordinator.

Notes

2.3 Position Turnover

[1] IF you are being relieved at the end of your shift or your position is otherwise being turned over, **THEN** perform a formal turnover as follows:

(a) Review status boards, WebEOC facility log, and other information with the position replacement as appropriate.

(b) Provide a briefing on the emergency, radiological conditions, and any actions that have been completed or are in progress.

(c) Inform the position's immediate supervisory position and affected staff that you are transferring responsibility for the position to the replacement individual.

(d) Sign off at the facility roster board, ensure your replacement signs in, and turn over the position binder and facility badge to the new position replacement.

3.1 Facility Deactivation

3.2 Termination

[1] Return the work area to its pre-emergency condition.

[2] Gather all logs, records, and documents and turn them over to the Engineering Coordinator.

[3] Notify the Engineering Coordinator when your position is deactivated.

ATTACHMENT 9.8 **OPERATIONS COORDINATOR**

Sheet 1 of 4

Name: ______________________________ Date:__________

Operations Coordinator

NOTE
Steps may be performed concurrently or out of sequence as deemed appropriate.

1.1 Actions For Initial Response

Notes

1.2 Facility Activation

[1] Report to the TSC and sign in on the TSC roster board.

[2] Begin and maintain a log of your work activities.

[3] Place the TSC ventilation system in the appropriate operational mode in accordance with Attachment 9.15, TSC/OSC Equipment Operation, and PNPS 2.2.134, "*TSC - HVAC System Normal and Postaccident Operation*." Assign task to OSC - Operations Support personnel if available. If not, assign Operations Engineer.

[4] Obtain briefing from the TSC Manager to coordinate Operations Engineering initial activation and assessment activities, which includes staffing needs, based on the nature of the emergency event.

NOTE
Prioritization of staffing positions should be based on the nature of the emergency and current and projected emergency response work activities.

[5] Ensure that the Operations Engineering staff is manned with Operations Engineers as needed.

[6] Assign a staff member to establish an open line of communications with the Control Room on the Mitigation Line.

[7] Ensure adequate staffing of the Control Room.

ATTACHMENT 9.8 **OPERATIONS COORDINATOR**

Sheet 2 of 4

Notes

2.1 Facility Operation

2.2 Operation

[1] Assist Operations-related engineering assessment as requested.

[2] Assist the Work Control Coordinator in determining the priority assigned to OSC activities.

[3] Provide up-to-date plant status to the TSC Manager and the TSC Communicator in order to maintain WebEOC, facility logs, and status boards current.

[4] Maintain an open line of communications with the Control Room over the Mitigation Line.

(a) Use a copy of the Essential Information Checklist from EP-IP-100, *"Emergency Classification and Notification"*, to gather information regarding the status of the event and provide to the EPM.

[5] Provide technical assistance to the Shift Manager.

[6] Ensure current emergency classification and EAL are correct.

[7] Coordinate Operations activities outside of the Control Room with the TSC Manager.

[8] Continually assess actual and projected plant status in relation to the Emergency Action Levels (EP-IP-100, *"Emergency Classification and Notification"*) and make recommendations to the TSC Manager on changes in the emergency classification.

ATTACHMENT 9.8 **OPERATIONS COORDINATOR**

Sheet 3 of 4

Notes

[9] Upon entry into EOPs, initiate EP-IP-225, *"Severe Accident Management Support"*, to support EOP implementation and accident mitigation strategies as needed.

(a) At the direction of the EPM, assume the duties and responsibilities of the Evaluator, or Decision-Maker if qualified, when transition to Severe Accident Management Guidelines (SAMG/SAG) is initiated.

[10] Approve emergency special procedures and implement as required under the provisions of 10CFR50.54(x).

[11] Provide technical assistance to the Shift Manager.

[12] Direct the activities of the Operations Engineering staff as needed.

Notes

2.3 Position Turnover

[1] **IF** you are being relieved at the end of your shift or your position is otherwise being turned over, **THEN** perform a formal turnover as follows:

(a) Review status boards, WebEOC facility log, and other information with the position replacement as appropriate.

(b) Provide a briefing on the emergency, radiological conditions, and any actions that have been completed or are in progress.

(c) Inform the position's immediate supervisory position and affected staff that you are transferring responsibility for the position to the replacement individual.

(d) Sign off at the facility roster board, ensure your replacement signs in, and turn over the position binder and facility badge to the new position replacement.

3.1 Facility Deactivation

3.2 Termination

[1] Return the work area to its pre-emergency condition.

[2] Secure the TSC/OSC ventilation system in accordance with Attachment 9.15, TSC/OSC Equipment Operation, and PNPS 2.2.134, *"TSC - HVAC System Normal and Postaccident Operation"*, as appropriate.

[3] Gather all logs, records, and documents and turn them over to the TSC Manager.

[4] Notify the TSC Manager when your position is deactivated.

ATTACHMENT 9.9 **TSC ENGINEERS (MECH, ELEC, I&C, OPS)**

Sheet 1 of 3

Name: ______________________________ Date:__________

TSC Engineer

NOTE

Steps may be performed concurrently or out of sequence as deemed appropriate.

Notes

1.1 Actions For Initial Response

1.2 Facility Activation

[1] Report to the TSC and sign in on the TSC roster board.

[2] Begin and maintain a log of your work activities.

[3] Obtain briefing from the Engineering Coordinator to determine status of plant and current priorities.

2.1 Facility Operation

2.2 Operation

[1] Perform the technical support, engineering activities and/or actions to be taken to mitigate plant damage in accordance with the priorities established by the Engineering/Operations Coordinator.

(a) Receive and update TSC engineering tasks using the Emergency Action Item Tracking Form via WebEOC, or if WebEOC is not available, using Attachment 9.19, Emergency Action Item Tracking Form.

[2] Use available computer systems along with communications with the Control Room to monitor and assess vital plant parameters and conditions.

ATTACHMENT 9.9 **TSC ENGINEERS (MECH, ELEC, I&C, OPS)**

Sheet 2 of 3

Notes

[3] Trend and assess plant parameters and status to:

(a) Determine the condition of safety-related systems and the fission product barriers.

(b) Verify that the status of equipment out of service is maintained.

(c) Provide recommendations for mitigating activities.

(d) Forecast expected changes in the level of plant and system safety.

(e) Determine the extent of core damage.

[4] **IF** requested, **THEN** perform monitoring, assessment, and evaluation in accordance with EP-IP-225, *"Severe Accident Management Support"*, to support EOP implementation and accident mitigation strategies as needed.

[5] **IF** requested, **THEN** develop or modify Procedures to perform response activities as necessary (such as emergency repairs or emergency system lineups).

[6] Provide engineering support for OSC activities as requested.

Notes

2.3 Position Turnover

[1] **IF** you are being relieved at the end of your shift or your position is otherwise being turned over, **THEN** perform a formal turnover as follows:

(a) Review status boards, WebEOC facility log, and other information with the position replacement as appropriate.

(b) Provide a briefing on the emergency, radiological conditions, and any actions that have been completed or are in progress.

(c) Inform the position's immediate supervisory position and affected staff that you are transferring responsibility for the position to the replacement individual.

(d) Sign off at the facility roster board, ensure your replacement signs in, and turn over the position binder and facility badge to the new position replacement.

3.1 Facility Deactivation

3.2 Termination

[1] Return the work area to its pre-emergency condition.

[2] Gather all logs, records, and documents and turn them over to the Engineering Coordinator.

[3] Notify the Engineering Coordinator when your position is deactivated.

ATTACHMENT 9.10 **TSC IT SPECIALIST**

Sheet 1 of 2

Name: ______________________________ Date:__________

TSC IT Specialist

NOTE
Steps may be performed concurrently or out of sequence as deemed appropriate.

1.1 Actions For Initial Response

Notes

1.2 Facility Activation

[1] Report to the TSC and sign in on the TSC roster board.

[2] Begin and maintain a log of your work activities.

[3] Report to the Engineering Coordinator for instructions.

2.1 Facility Operation

2.2 Operation

[1] Ensure activation and operation of TSC equipment and systems in accordance with Attachment 9.15, TSC/OSC Equipment Operation, except TSC ventilation which is attended to by the Operations Coordinator.

[2] Ensure that the WebEOC is activated and functional in accordance with the locally posted instructions.

[3] Assist in the overall operation of the TSC and TSC personnel as necessary.

Notes

2.3 Position Turnover

[1] **IF** you are being relieved at the end of your shift or your position is otherwise being turned over, **THEN** perform a formal turnover as follows:

(a) Review status boards, WebEOC facility log, and other information with the position replacement as appropriate.

(b) Provide a briefing on the emergency, radiological conditions, and any actions that have been completed or are in progress.

(c) Inform the position's immediate supervisory position and affected staff that you are transferring responsibility for the position to the replacement individual.

(d) Sign off at the facility roster board, ensure your replacement signs in, and turn over the position binder and facility badge to the new position replacement.

3.1 Facility Deactivation

3.2 Termination

[1] Return the work area to its pre-emergency condition.

[2] Gather all logs, records, and documents and turn them over to the Engineering Coordinator.

[3] Notify the Engineering Coordinator when your position is deactivated.

1.1 Emergency Task Assignment Directions

1.2 Task Assignments

NOTE

WebEOC should be used to open a new record from the Repair/Corrective Action (Admin) menu to initiate assignments to OSC teams.

[1] A task is identified and initiated by any member of the ERO and documented on the Emergency Task Assignment Sheet and Directions form (Sheet 6 of this Attachment). The first section of the form is completed as follows:

(a) Subject: A title or brief description of the task.

(b) Description: A description of the task objective. This section is also used to identify individual subtasks in an outline fashion for tasks involving multiple activities.

(1) Additional subtasks may also be identified and added to the task description following Emergency Plant Manager approval (the Emergency Task Assignment Sheet and Directions form is a working document).

(2) Whenever subtasks are added after Emergency Plant Manager approval, the Emergency Plant Manager must be informed and the activity listed on the tracking form/status board.

(c) Originator: Name of individual originating task.

(d) Estimated Duration: Estimated duration of the task.

(e) Assigned To: Denotes the facility(ies), priority, and support groups needed.

ATTACHMENT 9.11 **EMERGENCY TASK ASSIGNMENT SHEET AND DIRECTIONS**

Sheet 2 of 7

(f) Instruction/Results: Record key information and instructions considering the items below:

(1) Nature of the task including applicable Procedures, if available.

(2) Equipment, tools, instrumentation, and materials necessary for the task.

(3) Physical location where the task is performed, including system and equipment numbers as applicable.

(4) Safety precautions pertaining to both the task and to personnel.

(5) Communications equipment, channels, backup, and reporting expectations.

(6) Any special instructions applicable to the task or evolution.

[2] The task is submitted through the appropriate ERO managers to the Emergency Plant Manager for approval and logging.

(a) Tasks directed from the Control Room (over the Mitigation Line or other communications system) are generated and documented by the off-shift SRO in the TSC.

(b) Tasks originating from within the technical or Engineering support groups are provided to the Engineering Coordinator.

(c) Tasks originating from within the Operations support group are provided to the Operations Coordinator.

(d) Tasks originating from other parts or levels of the ERO are provided directly to the Emergency Plant Manager.

ATTACHMENT 9.11 **EMERGENCY TASK ASSIGNMENT SHEET AND DIRECTIONS**

Sheet 3 of 7

[3] Prior to approval of a task involving the OSC, the Emergency Plant Manager shall consult with other key emergency response personnel and discuss the following (as applicable):

(a) Sequencing and logistics for accomplishing the task. Ensure that enough technical support is provided to the OSC Manager and the Radiological Coordinator for tasks involving the OSC.

(b) Existing or potential hazards to personnel (for example: oxygen levels, explosive atmosphere, electrical, steam, obstructions, toxic substances).

(c) Time constraints for performance of the task activities.

[4] Tasks will be given a unique job identification number by WebEOC to allow tracking and prioritization of plant support functions:

(a) Additional subtasks can be added as the activities necessary to the completion of the task are identified. Emergency Plant Manager approval is only required for each task, not for each subtask.

[5] Tasks are then distributed as follows:

(a) TSC tasks are provided to the Engineering Coordinator for tracking and resolution/completion.

> NOTE
>
> To ensure that OSC activities are conducted only under carefully controlled and preplanned conditions, briefing forms are used. However, if it is determined that completion of these forms will impede a timely response, teams may be briefed and dispatched prior to completion of the documentation.

(b) OSC tasks are provided to the Work Control Coordinator for tracking and resolution/completion and to the Radiological Coordinator for the radiological controls determination.

ATTACHMENT 9.11 **EMERGENCY TASK ASSIGNMENT SHEET AND DIRECTIONS**

Sheet 4 of 7

[6] Once issued, OSC activities are prioritized and tracked by the Work Control Coordinator on the Emergency Task Tracking Form (Attachment 9.12) and WebEOC or by similar method (such as status board) to provide a dynamic illustrated overview of support functions conducted during the emergency. The Engineering Coordinator prioritizes and tracks TSC tasks.

(a) Priorities are determined as being High (1), Medium (2), or Low (3) as follows:

- High (1): The task is necessary to protect the immediate health and safety of the public. High priority tasks are in response to plant conditions that are allowing the rapid deterioration of safety barriers, or barriers have already been broken such that a release is either occurring or imminent.
- Medium (2): Any task that requires action by the TSC/OSC and should be worked on in the immediate time period but does not fit the criteria of a health and safety of the public related item (for example, if a system has only one remaining component, repair of the backup components).
- Low (3): Any task which can be worked on when resources permit (i.e., getting meals, preparations for recovery activities).

1.3 TSC Tasks

[1] Tasks provided to the Work Control Coordinator are reviewed to determine the applicable support area(s) to be assigned task responsibility and to maintain control of the prioritization of TSC activities.

[2] ERO personnel assigned responsibility for the task will:

(a) Provide the requested support.

(b) Document the results as appropriate. Additional pages may be added to the Emergency Task Assignment Sheet and Directions form to describe newly identified subtasks or further discuss results. The objective of the task documentation is to provide enough information to allow reconstruction of events and historical information to relief personnel, not to chronicle the activity in the detail of an incident report to the detriment of necessary support functions.

(c) Close out the task, when completed, by filling out the Completed By and Time blocks at the bottom of the last page of the Emergency Task Assignment Sheet and Directions form and returning the package to the Emergency Plant Manager.

(d) Provide status updates to the TSC Manager as appropriate.

[3] The Engineering Coordinator will maintain the Engineering Action Item Tracking Forms (Attachment 9.19)/status boards up-to-date and periodically brief the TSC Manager and the Emergency Plant Manager on the status of TSC activities.

ATTACHMENT 9.11 EMERGENCY TASK ASSIGNMENT SHEET AND DIRECTIONS

Sheet 6 of 7

TASK	Page 1 of ___
Subject:	
Description:	
Originator:	**Estimated Duration:**

ASSIGNED TO	
Facility: **Priority:**	**Support Groups Needed:**
☐ TSC	☐ Operations ☐ Systems
☐ OSC	☐ Security ☐ Maintenance
☐ Other (specify:________________)	☐ Radiological ☐ Chemistry
	☐ Materials ☐ Other_______

ASSIGNMENT AND APPROVAL		
EPM Approval:	**Task No:**	**Time:**

ATTACHMENT 9.11 **EMERGENCY TASK ASSIGNMENT SHEET AND DIRECTIONS**

Sheet 7 of 7

INSTRUCTIONS/RESULTS	
Discussion:	
Completed By:	**Time:**

ATTACHMENT 9.12 **EMERGENCY TASK TRACKING FORM**

Sheet 1 of 2

OSC Activities

No.	Task	Condition	Priority	Status/Resolution
		Forming Dispatched Completed		

ATTACHMENT 9.12 **EMERGENCY TASK TRACKING FORM**

Sheet 2 of 2

TSC Activities

No.	Task	Condition	Priority	Status/Resolution
		Working Completed		

AREA RADIATION MONITORS

TIME:

PANEL/ID NO.	MONITOR	TREND	READING		ALARM IN	RANGE
C910/1705-60	Carbon Bed Vault Area			mR/Hr		$10^{-1} - 10^{3}$
C911/1815-3A	Condensate Pump Stairway			mR/Hr		$10^{-1} - 10^{3}$
C911/1815-8A	Feedwater Heaters			mR/Hr		$10^{0} - 10^{4}$
C911/1815-2A	Main Control Room			mR/Hr		$10^{-2} - 10^{2}$
C911/1815-8B	Turbine-Front Standard			mR/Hr		$10^{0} - 10^{4}$
C911/1815-3B	Radwaste-Corridor			mR/Hr		$10^{-1} - 10^{3}$
C911/1815-8C	Radwaste-Sump Area			mR/Hr		$10^{0} - 10^{4}$
C911/1815-8D	Chem. Waste Tank			mR/Hr		$10^{0} - 10^{4}$
C911/1815-2B	Rx-Outside TIP Room			mR/Hr		$10^{-2} - 10^{2}$
C911/1815-2C	Radwaste Shipping Lock			mR/Hr		$10^{-2} - 10^{2}$
C911/1815-2D	Rx Access Area (S.E.)			mR/Hr		$10^{-2} - 10^{2}$
C911/1815-3C	New Fuel Storage Area			mR/Hr		$10^{-1} - 10^{3}$
C911/1815-3D	New Fuel Vault			mR/Hr		$10^{-1} - 10^{3}$
C911/1815-3E	Shield Plug Area			mR/Hr		$10^{-1} - 10^{3}$
C911/1815-3F	Spent Fuel Pool Area			mR/Hr		$10^{-1} - 10^{3}$

1705-60} 500mR/Hr	1815-8D} 300 mR/Hr
1815-3A} 55 mR/Hr	1815-2B} 5 mR/Hr
1815-8A} 600 mR/Hr	1815-2C} 15 mR/Hr
1815-2A} 1 mR/Hr	1815-2D} 60 mR/Hr
1815-8B} 400mR/Hr	1815-3C} 6 mR/Hr
1815-3B} 15 mR/Hr	1815-3D} 40 mR/Hr
1815-8C} 6000 mR/Hr	1815-3E} 40 mR/Hr
	1815-3F} 40 mR/Hr

OOS – Out of Service OSH -- Off Scale Hi DS – Down Scale

Circle readings that are greater than alarm setpoint in red.

ATTACHMENT 9.14 **PROCESS RADIATION MONITOR DATA FORM**

Sheet 1 of 1

PANEL/ID NO.	MONITOR	TREND	READING		ALARM IN	RANGE
C910/1705-18	Main Stack Lo A			CPS		10^{-1} -
10^{6}						
	Main Stack Lo B			CPS		10^{-1} - 10^{6}
C910/1001-608	Main Stack Hi			R/Hr		10^{-1} - 10^{4}
C910/1705-32	Rx Bldg Vent Lo A			CPS		10^{-1} - 10^{6}
	Rx Bldg Vent Lo B			CPS		10^{-1} - 10^{6}
C910/1001-609	Rx Bldg Vent Hi			R/Hr		10^{-1} - 10^{4}
C910/1001-610	Turbine Bldg Vent Hi			R/Hr		10^{-1} - 10^{4}
C170/1001-606A	Drywell CHRMS A			R/Hr		10^{0} - 10^{7}
C171/1001-606B	Drywell CHRMS B			R/Hr		10^{0} - 10^{7}
C170/1001-607A North	Torus CHRMS A			R/Hr		10^{0} - 10^{7}
C171/1001-607B East	Torus CHRMS B			R/Hr		10^{0} - 10^{7}
C910/1705-2	Main Steam Line A			mR/Hr		10^{0} - 10^{6}
	Main Steam Line B			mR/Hr		10^{0} - 10^{6}
	Main Steam Line C			mR/Hr		10^{0} - 10^{6}
	Main Steam Line D			mR/Hr		10^{0} - 10^{6}
C910/1705-3	Air Ejector Off Gas A			mR/Hr		10^{0} - 10^{6}
	Air Ejector Off Gas B			mR/Hr		10^{0} - 10^{6}
C910/1705-4	A Loop RBCCW			CPS		10^{-1} - 10^{6}
	B Loop RBCCW			CPS		10^{-1} - 10^{6}
C910/1705-8	Refuel Floor Vent A			mR/Hr		10^{-1} - 10^{3}
	Refuel Floor Vent B			mR/Hr		10^{-1} - 10^{3}
	Refuel Floor Vent C			mR/Hr		10^{-1} - 10^{3}
	Refuel Floor Vent D			mR/Hr		10^{-1} - 10^{3}
C910/1705-9	SBGT Exhaust			mR/Hr		10^{0} - 10^{4}
C910/1705-16	Control Rm Air Intake			mR/Hr		10^{-2} - 10^{2}
C910/1705-30	R/W Effluent Discharge			CPS		10^{-1} - 10^{6}
C910/1705-5	Offgas Post Treatment A			CPS		10^{-1} - 10^{6}
	Offgas Post Treatment B			CPS		10^{-1} - 10^{6}

FLOW RATES Panel C7

SBGT FL8126/7 ______________(CFM)

Rx Bldg FL8116A ______________(CFM)

(CIRCLE) Normal / Isolated

Main Stack __________ CFM*

TB Vent Exh. __________ CFM*

TB Roof Fans __________ Running*

MET DATA Panel MT1

Delta Temp __________Deg. F

Outside Temp __________Deg. F

Stability Class __________

	220' Met Tower		160' Met Tower*	
	Upper (220')	Lower (33')	Upper (160')	Lower (33')
Dir (from)	__Deg	_Deg______	Deg______	_Deg
Speed	________mph	_mph______	_mph______	_mph

*Not provided from Control Room

ATTACHMENT 9.15 **TSC/OSC EQUIPMENT OPERATION**

Sheet 1 of 5

1.0 TSC/OSC Ventilation System Operation

[1] Ensure all TSC boundary doors (including the air lock doors) are not propped open or otherwise impeded from fully closing (see Attachment 9.16, TSC Pressure Boundary/Location Map).

> NOTE
>
> TSC/OSC ventilation system operation is to be conducted/coordinated by the Operations Coordinator or designee only.

[2] Place TSC ventilation system in the appropriate operational mode in accordance with PNPS 2.2.134, "*TSC - HVAC System Normal and Postaccident Operation.*"

2.0 Fax Machine Operation

[1] Locate the fax machine on the table adjacent to the door leading to the TSC Operations area.

[2] In the backside of the fax machine should be a telephone cable plugged into the slot labeled "LINE." Ensure the other end of the telephone cable is plugged into the phone jack marked “FAX”. Check dial tone.

[3] After the fax machine has been set up and turned on, follow the operating instructions on the slideout plate underneath the fax machine to transmit documents.

3.0 Public-Address System Operation

[1] Locate the controls for the public-address system in the Communication Cabinet against the partition wall separating the TSC/OSC.

[2] Set amplifier power switch to "ON".

[3] Set EPM dial for the appropriate volume level. Set the unused dials to zero.

[4] Set optional page zone switches (remote stereo speakers) to "ON."

(a) Zone Switch #1 = Bullpen, Back Area

(b) Zone Switch #2 = Computer Room

(c) Zone Switch #3 = Conference Room

[5] Turn on the microphone located on the Emergency Plant Manager's desk.

[6] A wireless microphone is located on top of the public-address system amplifier.

4.0 Mitigation Line Operation

[1] There are three extensions of the Mitigation Line in the TSC:

(a) Operations area

(b) Emergency Plant Manager's (EPM) desk

(c) TSC Manager's desk

[2] Locate the desired Mitigation Line telephone.

[3] Operate in accordance with the instructions that are posted at each Mitigation Line telephone location or see "Mitigation Line Network Diagram" in Emergency Telephone Directory, Section 6, Communication Systems Instructions/Diagrams.

[4] To use the speaker phone, press speaker phone "SPKR" button.

5.0 Plant Data Phone (PDP) Operation

[1] There are two extensions of the PDP in the TSC:

(a) On wall between plant parameter status boards.

(b) Corner desk in the Operations area.

[2] Locate the desired Plant Data Phone (brown).

[3] Operate in accordance with the instructions that are posted at each PDP location or see "Plant Data Phone (PDP) Diagram" in Emergency Telephone Directory, Section 6.

6.0 Emergency Conference Link (ECL) Operation

[1] There are two extensions of the ECL in the TSC:

(a) Emergency Plant Manager's desk.

(b) TSC Conference Room.

[2] Locate the desired Emergency Conference Link telephone.

[3] Operate in accordance with the instructions that are posted at each ECL location attached to the ECL or see "Emergency Conference Line (ECL) Diagram" in Emergency Telephone Directory, Section 6.

7.0 Core Damage Computer

[1] The core damage computer code can be accessed by simply energizing (On/Off switch) the computer.

[2] For detailed instructions on operation, refer to EP-IP-330, *"Core Damage"*.

8.0 Dosimetry Database (DDB) Terminal

The DDB terminal located in the respirator fit room is a menu-driven system that can be accessed by energizing (On/Off switch) the terminal and logging onto the system using your password.

9.0 EPIC Workstations/SPDS

Emergency Plant Information Computer (EPIC) workstations allow access to plant data and Safety Parameter Display System (SPDS) displays.

[1] Ensure the EPIC workstations are operational.

10.0 Business Network Computers, Monitors and Printers

Business network computers are provided in the TSC/OSC area for access to network software. Individuals using these computers should use their own logon/password combination to gain access to network software.

[1] Open the Business Network Computer Cabinet in the Operations Area. Ensure that the business network computer is available for use.

[2] Ensure that the other business network computers, monitors and printers are available for use, including the monitor in the OSC Pool Area.

11.0 Disaster Recovery Computer and Printer

[1] Ensure that the Disaster Recovery Computer and printer in the Operations Area are available for use.

12.0 Emergency Response Data System (ERDS) Verification/Activation (if requested by Control Room)

[1] The ERDS transmits Station parameters to the NRC for use in a declared emergency. The PNPS ERDS is in continuous operation.

(a) Log on to the Entergy business network at a TSC computer, preferably at the Business Network Computer Cabinet or Engineering Coordinator computer because the ERDS application is already installed.

(b) Click Start - All Programs - Nuclear Corporate Applications (ESM) - ERDS Activation - Site Activation Display.

(c) Click on "Continue" when the "Warning" display screen appears.

(d) Click the "View Only" button to enter the View Only mode.

(e) Verify that a Transmitting Data message appears in the ERDS status field and that the number in the Messages Sent field is incrementing upwards. If not, perform the following:

(1) Exit the ERDS application.

(2) Click Start - All Programs - Nuclear Corporate Applications (ESM) - ERDS Activation - Site Activation Display.

(3) Click on "Continue" when the "Warning" display screen appears.

(4) Enter the appropriate pass code from Section 2 of the Emergency Telephone Directory on the pass code entry screen and press the "Submit" button.

(5) Click the "Connect" button on the PNP ERDS display to activate ERDS.

Record ERDS activation date/time:________________/______________

(6) Verify that a Transmitting Data message appears in the ERDS status field and that the number in the Messages Sent field is incrementing upwards. If not, resolve as necessary.

(f) At the end of the event, exit the ERDS application. Do not disconnect ERDS.

13.0 Copy Machine

[1] Ensure the copy machine is turned on and functional.

14.0 TSC UPS (Uninterruptible Power Supply) System

Operation and maintenance of the TSC UPS system is described in PNPS 3.M.3-22, *"TSC Uninterruptible Power Supply System Maintenance"*, and PNPS 3.M.3-23, *"TSC Electrical Diesel System Maintenance"*.

15.0 Securing TSC/OSC Ventilation System Operation

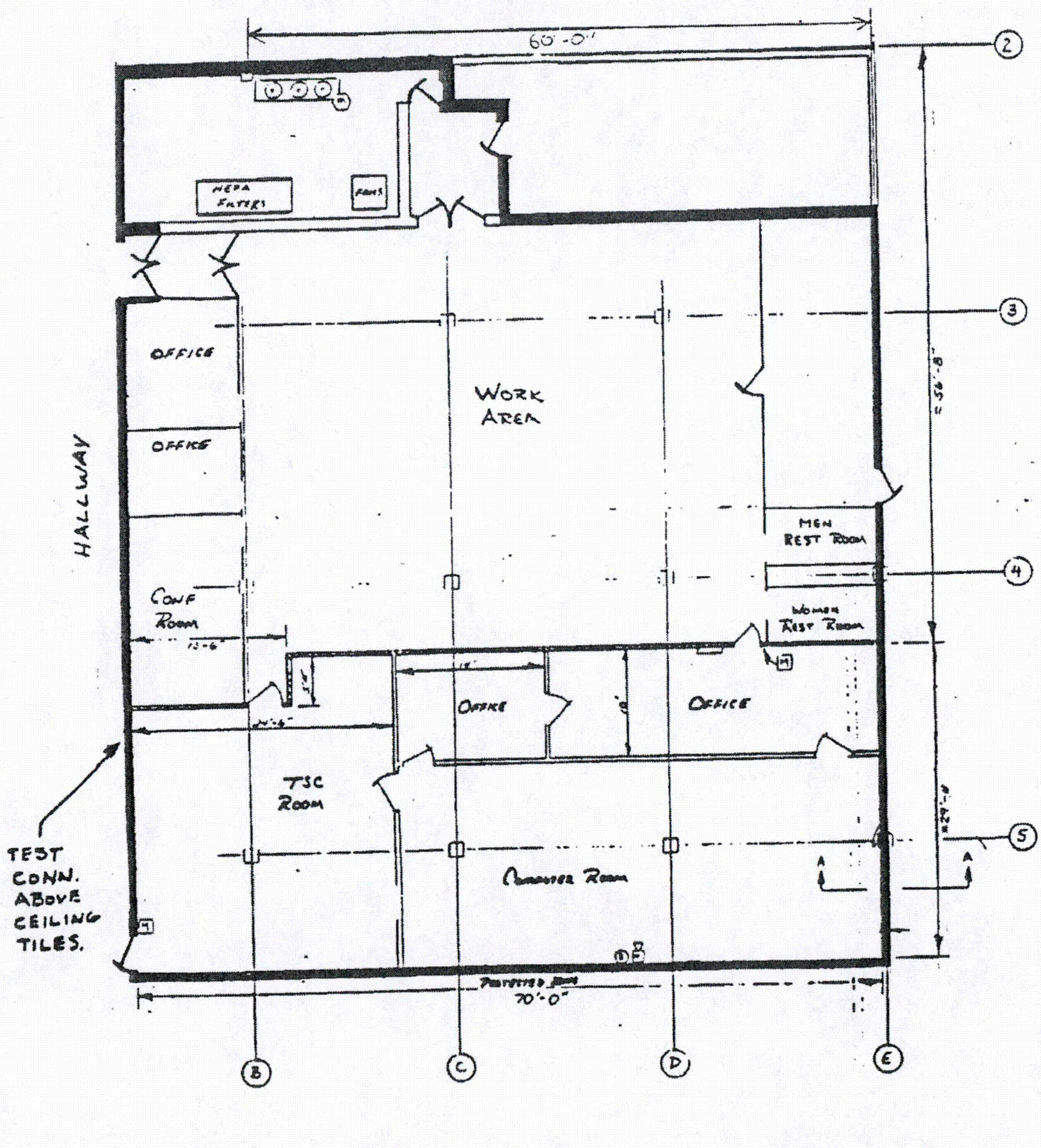
NOTE
TSC/OSC ventilation system operation is to be conducted/coordinated by the Operations Coordinator or designee only.

Perform the following step to return to normal service **ONLY IF** a postaccident condition **DOES NOT** exist.

[1] Secure the TSC ventilation system in accordance with PNPS 2.2.134, *"TSC - HVAC System Normal and Postaccident Operation"*.

ATTACHMENT 9.16 **TSC PRESSURE BOUNDARY/LOCATION MAP**

Sheet 1 of 1



ATTACHMENT 9.17 — **PLANT DATA FORM**

Sheet 1 of 1

1 AC POWER SOURCES

UAT	insrvc avail oos
S/U XFMR	insrvc avail oos
S/D XFMR	insrvc avail oos
EDG A	insrvc avail oos
EDG B	insrvc avail oos
BODG	insrvc avail oos

2 SAFEGUARDS

Rx Trip Signal___________

Mode Switch	S/D SU/HSB	Refuel Run

SLC Pumps	A	insrvc avail oos
	B	insrvc avail oos

SLC TK LVL_________gal

3 RHR — Mode

Loop A: A insrvc avail oos — Torus Cooling
C insrvc avail oos — DW Spray
Torus Spray
LPCI
Loop Flow_______GPM — SDC

Loop B: B insrvc avail oos — Torus Cooling
D insrvc avail oos — DW Spray
Torus Spray
LPCI
Loop Flow_______GPM — SDC

4 CS

Loop A:
insrvc avail oos
Loop Flow:
_______GPM

Loop B:
insrvc avail oos
Loop Flow:
_______GPM

5 PCIS

	Required	Isolated	
GP 1	Y N	Y N	(MSIVs)
GP 2	Y N	Y N	(Pri Cont)
GP 3	Y N	Y N	(SDC)
GP 4	Y N	Y N	(HPCI)
GP 5	Y N	Y N	(RCIC)
GP 6	Y N	Y N	(RWCU)

Bypassed_________________

6 COOLING WATER

RBCCW Pumps	Loop A	A	insrvc avail oos
		B	insrvc avail oos
		C	insrvc avail oos
	Loop B	D	insrvc avail oos
		E	insrvc avail oos
		F	insrvc avail oos
TBCCW Pumps		A	insrvc avail oos
		B	insrvc avail oos

SSW Pumps	Loop A	A	insrvc avail oos
		B	insrvc avail oos
	Loop B	D	insrvc avail oos
		E	insrvc avail oos
	Swing	C	insrvc avail oos
Seawater Pumps		A	insrvc avail oos
		B	insrvc avail oos

7 HPCI

insrvc avail oos
D Pressure Control
D Level Control

8 RCIC

insrvc avail oos
D Pressure Control
D Level Control

9 RECIRC

Recirc Pumps	A insrvc avail oos
	______% speed
Recirc Pumps	B insrvc avail oos
	______% speed

10 CRD

CRD Pumps	A	insrvc avail oos
	B	insrvc avail oos

11 SGT

A	insrvc avail oos
B	insrvc avail oos

12 FEED/COND

Cond Pumps	A	insrvc avail oos
	B	insrvc avail oos
	C	insrvc avail oos
Feed Pumps	A	insrvc avail oos
	B	insrvc avail oos
	C	insrvc avail oos
CST Level	A______ft	
	B______ft	

13 RB VENT

insrvc isolated
RB d/p_____inches H_2O

CRITICAL PARAMETERS

TIME			TREND ↑←→↓
REACTOR POWER	%		
RPV PRESSURE	psig		
RPV WATER LEVEL	inches		
DRYWELL TEMP	deg F		
TORUS WATER TEMP	deg F		
TORUS WATER LEVEL	inches		
DRYWELL PRESSURE	psig		
TORUS BOTTOM PRESSURE	psig		
CONTAIN. H_2 CONC.	%		
CONTAIN. O_2 CONC.	%		

ATTACHMENT 9.18 **TSC BRIEFING GUIDELINE - PERIODIC BRIEF**

Sheet 1 of 3

PERIODIC BRIEF

NOTE

When identifying critical parameters to be monitored, be sure to identify who owns the parameter, by name, and their expected actions upon changes in that parameter.

I. Prior to brief, coordinate updates from key positions:

- ☐ Engineering Coordinator: Review and status of engineering priorities.
- ☐ Operations Coordinator: Changes in key plant parameters, any significant feedback.
- ☐ TSC Communicator: Significant communications problems or issues, time and status of upcoming notifications (if applicable).
- ☐ Radiological Coordinator: Current radiological conditions in plant, radiological release status, field team radiological exposure concerns, TSC habitability status.
- ☐ TSC Security Coordinator: Security status, evacuation status (if applicable).

ATTACHMENT 9.18 **TSC BRIEFING GUIDELINE - PERIODIC BRIEF**

Sheet 2 of 3

II. Emergency Plant Manager:

☐ About 10 minutes ahead, announce, "Next TSC briefing at_____________(time)."

☐ To start the brief, announce *"Time for TSC briefing."* Obtain attention of team.

☐ Emergency Declaration is______________at _______________ based upon EAL _________________

☐ Reactor Status:

Shutdown ☐ Yes at ______________ (Time) ☐ No

At: ______________ (Time)

Rx Power __________ Rx Pressure __________ Rx Water Level __________

☐ Factors that may require escalation include:______________________________

____

☐ Plant Status (significant parameters):______________________________

____

☐ TSC Priorities:____

____

III. **Emergency Plant Manager** - Provide during "Initial Brief" as needed:

a. Need to keep a questioning attitude

b. Use the STAR process

c. Use peer-checks

d. Use of 3-part communication at all times

e. Placekeeping (stay focused)

f. Use of Procedures

g. Safety - Maintain highest margin

h. Full attention for facility briefs

i. Stay alert - attention to details

j. Use the 30-minute rule - Check with teams (OSC, Control Room, etc.) at least every 30 minutes to keep abreast of activities.

k. Professionalism

l. Maintain logbooks up-to-date

m. Questioning attitude

n. Anyone leaving the TSC Envelope must be briefed by the Radiological Coordinator.

IV. **Emergency Plant Manager** - Briefing Wrap-up:

a. Final review of TSC priorities

b. Questions/Comments

"End of Brief"

ATTACHMENT 9.19 **ENGINEERING ACTION ITEM TRACKING FORM**

Sheet 1 of 1

ENGINEERING ACTION ITEM TRACKING		
Entered By		
Requested By		
Requested Date/Time		(MM/DD/YYYY HH:MM)
Assigned To		
Status	Assigned	
	Completed (Successful)	
	Completed (Unsuccessful)	
	Rejected	
	Requested	
Action Item Description		
Notes		
Date/Time Assigned		(MM/DD/YYYY HH:MM)
Prioritization	High	
	Medium	
	Low	
Date/Time Due		(MM/DD/YYYY HH:MM)
Date/Time Completed		(MM/DD/YYYY HH:MM)
Completion Comments		

ATTACHMENT 9.20 FILLING AN ERO VACANCY DURING FACILITY ACTIVATION/OPERATION

Sheet 1 of 2

[1] If an ERO position is not filled during the ERF activation/operation, the facility manager (if possible, a non-on-call facility manager) should take the following action to fill the vacancy to ensure the facility can perform its assigned function and the facility can be made operational within the time requirements established in the PNPS Emergency Plan.

NOTE

Vacant positions can be filled using ERO members from other emergency response facilities.

- If possible, select a person to fill the position with any individual qualified for that position. This person does not need to be the on-duty person. In an all-call activation, the first responder to arrive for a position should take the position until the on-duty person arrives.
- If possible, fill the position with a person previously qualified for the vacant position. This fill-in person may have a new ERO position for which there is adequate ERO staffing.
- If possible, select a person from the extra ERO members reporting that are responsible for performing the facility function (i.e., dose assessment, communication, administrative support, etc.) as the vacant position.
- If possible, the person selected should be from a higher position in the reporting chain of the vacant position (i.e., Engineering Coordinator to fill for a Mechanical/Electrical/I&C Engineer, Emergency Plant Manager to fill for a TSC Manager, EOF Manager to fill for an EOF Communicator, etc.).
- If none of the above steps are possible, fill the position with a person who is technically qualified to perform the assigned task and can use the position book, checklist, etc., as their guide. As an example, a Maintenance supervisor not yet qualified in the ERO could be assigned as the Mechanical Coordinator. Although this approach is the least desirable, the goal is to activate the respective ERF as soon as possible to support mitigation of the event.

ATTACHMENT 9.20 FILLING AN ERO VACANCY DURING FACILITY ACTIVATION/OPERATION

Sheet 2 of 2

[2] Once a person is selected to fill the vacant position, perform the following:

- Ensure the person filling the vacancy understands their new duties by having them review the position binder, checklist, etc.
- Ensure the person filling the vacancy is wearing the appropriate badge.
- Ensure the person filling the vacancy understands their new role in the ERO.
- Ensure the facility lead is aware of the actions taken to fill the vacancy.
- Contact the Administration and Logistics Coordinator in the EOF and instruct him/her to locate someone from the vacant position and have them report immediately to the facility.

[3] If the vacant ERO position is the facility manager position, then this responsibility would become the facility lead's responsibility. This responsibility can be delegated.

ATTACHMENT 9.21 **DOCUMENT CROSS-REFERENCE**

Sheet 1 of 1

This Attachment lists those documents, other than source documents, which may be affected by changes to this Procedure.

Document Number	Document Title
EP-IP-100	Emergency Classification and Notification
EP-IP-210	Control Room Augmentation
EP-IP-225	Severe Accident Management Support
EP-IP-260	Emergency Operations Facility (EOF) Operations
EP-IP-262	Operations Support Center (OSC) Operations
EP-IP-315	Personnel Monitoring Team Activation and Response
EP-IP-330	Core Damage
EP-IP-400	Protective Action Recommendations
EP-IP-410	Evacuation/Assembly
EP-IP-420	Search and Rescue
EP-IP-440	Emergency Exposure Controls
EP-AD-122	Maintenance of the Emergency Telephone Directory
EP-AD-601	Emergency Action Level Technical Bases Document
PNPS 1.3.12.1	Non-Emergency Notification of Management

Procedure/Document Number: EP-IP-261	**Revision:** 8
Equipment/Facility/Other: Pilgrim Nuclear Power Station	
Title: Technical Support Center (TSC) Operations	

Part I. Description of Activity Being Reviewed This procedure revision:

1. Change "TSC IT Specialists" to "TSC IT Specialist". Pages 2
2. Add step for the Radiological Coordinator to ensure that the Rad/Chem Coordinator is "Appropriately controlling ALL entrances to the TSC/OSC facility". Page 35
3. Change "OSC/TSC" to "TSC/OSC". Page 35
4. Add Emergency Telephone Directory, Section 6 title "Communication Systems Instructions/Diagrams". Page 72
5. Add business "Monitors and Printers" to Attachment 9.15, TSC/OSC Equipment Operation. Page 73
6. Add Step "Open the Business Network Computer Cabinet in the Operations Area. Ensure that the business network computer is available for use." Page 73
7. Change "Ensure the business network computers are available." to "Ensure that the other business network computers, monitors and printers are available for use, including the monitor in the OSC Pool Area." Page 73
8. Add Section "11.0 Disaster Recovery Computer and Printer" with step "[1] Ensure that the Disaster Recovery Computer and printer in the Operations Area are available for use." Page 74
9. Change "ERDS Cabinet" to "Business Network Computer Cabinet". Page 74
10. Move old Attachment 9.15 TSC UPS (Uninterruptible Power Supply) System Section 11.0 to 14.0. Page 75
11. Delete "EP File 1.6.4 'DAMAGE' version 2.0 Computer Application Verification, Validation and Documentation' 11/22/95" from Document Cross-Reference. Page 84

Part II. Activity Previously Reviewed?		
Is this activity fully bounded by an NRC approved 10 CFR 50.90 submittal or Alert and Notification System Design Report? If YES, identify bounding source document number/approval reference and ensure the basis for concluding the source document fully bounds the proposed change is documented below:	☐ YES 50.54(q)(3) Evaluation is NOT required. Enter justification below and complete Part VI.	☒ NO Continue to next part

Justification:

☐ Bounding document attached (optional)

Part III. Applicability of Other Regulatory Change Control Processes

APPLICABILITY CONCLUSION

☒ If there are no controlling change processes, continue the 50.54(q)(3) Screening.

☐ One or more controlling change processes are selected, however, some portion of the activity involves the emergency plan or affects the implementation of the emergency plan; continue the 50.54(q)(3) Screening for that portion of the activity. Identify the applicable controlling change processes below.

☐ One or more controlling change processes are selected and fully bounds all aspects of the activity. 50.54(q)(3) Evaluation is NOT required. Identify controlling change processes below and complete Part VI.

CONTROLLING CHANGE PROCESSES

10 CFR 50.54(q)

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Part IV. Editorial Change Is this activity an editorial or typographical change such as formatting, paragraph numbering, spelling, or punctuation that does not change intent? Yes, Items 1,3-5,9-11. **Justification:** Item 1 corrects a plurality typo for the TSC IT Specialist Attachment. Item 3 changes the order of TSC and OSC to the Pilgrim Station convention. Item 4 adds the title of the referenced Emergency Telephone Directory section. Item 5 clarifies that computers includes monitors and printers. Item 9 updates the name of the cabinet to reflect the computer currently housed inside it. Item 10 moves a section around for formatting purposes. Item 11 removes a cross-reference to a file for a retired software application. "NO" is checked because the procedure revision contains other changes that are not editorial.	☐ YES 50.54(q)(3) Evaluation is NOT required. Enter justification and complete Part VI.	☒ NO Continue to next part

Part V. Emergency Planning Element/Function Screen (Associated 10 CFR 50.47(b) planning standard function identified in brackets) Does this activity affect any of the following, including program elements from NUREG-0654/FEMA REP-1 Section II?	
1. Responsibility for emergency response is assigned. [1]	☐
2. The response organization has the staff to respond and to augment staff on a continuing basis (24/7 staffing) in accordance with the emergency plan. [1]	☐
3. The process ensures that on shift emergency response responsibilities are staffed and assigned. [2]	☐
4. The process for timely augmentation of onshift staff is established and maintained. [2]	☐
5. Arrangements for requesting and using off site assistance have been made. [3]	☐
6. State and local staff can be accommodated at the EOF in accordance with the emergency plan. [3]	☐
7. A standard scheme of emergency classification and action levels is in use. [4]	☐
8. Procedures for notification of State and local governmental agencies are capable of alerting them of the declared emergency within 15 minutes after declaration of an emergency and providing follow-up notifications. [5]	☐
9. Administrative and physical means have been established for alerting and providing prompt instructions to the public within the plume exposure pathway. [5]	☐
10. The public ANS meets the design requirements of FEMA-REP-10, Guide for Evaluation of Alert and Notification Systems for Nuclear Power Plants, or complies with the licensee's FEMA-approved ANS design report and supporting FEMA approval letter. [5]	☐
11. Systems are established for prompt communication among principal emergency response organizations. [6]	☐
12. Systems are established for prompt communication to emergency response personnel. [6]	☐
13. Emergency preparedness information is made available to the public on a periodic basis within the plume exposure pathway emergency planning zone (EPZ). [7]	☐
14. Coordinated dissemination of public information during emergencies is established. [7]	☐
15. Adequate facilities are maintained to support emergency response. [8]	☐

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16. Adequate equipment is maintained to support emergency response. [8]	☐
17. Methods, systems, and equipment for assessment of radioactive releases are in use. [9]	☐
18. A range of public PARs is available for implementation during emergencies. [10]	☐
19. Evacuation time estimates for the population located in the plume exposure pathway EPZ are available to support the formulation of PARs and have been provided to State and local governmental authorities. [10]	☐
20. A range of protective actions is available for plant emergency workers during emergencies, including those for hostile action events.[10]	☐
21. The resources for controlling radiological exposures for emergency workers are established. [11]	☐
22. Arrangements are made for medical services for contaminated, injured individuals. [12]	☐
23. Plans for recovery and reentry are developed. [13]	☐
24. A drill and exercise program (including radiological, medical, health physics and other program areas) is established. [14]	☐
25. Drills, exercises, and training evolutions that provide performance opportunities to develop, maintain, and demonstrate key skills are assessed via a formal critique process in order to identify weaknesses. [14]	☐
26. Identified weaknesses are corrected. [14]	☐
27. Training is provided to emergency responders. [15]	☐
28. Responsibility for emergency plan development and review is established. [16]	☐
29. Planners responsible for emergency plan development and maintenance are properly trained. [16]	☐

APPLICABILITY CONCLUSION

☑ If no Part V criteria are checked, a 50.54(q)(3) Evaluation is NOT required; document the basis for conclusion below and complete Part VI.

☐ If any Part V criteria are checked, complete Part VI and perform a 50.54(q)(3) Evaluation.

BASIS FOR CONCLUSION

Item 2: Radiological boundaries are posted in accordance with Radiological Protection procedures. This change adds an administrative task for the Radiological Coordinator to ensure that the Rad/Chem Coordinator is appropriately controlling ALL entrances to the TSC/OSC facility after an entrance was overlooked during an EP drill. This change does not change intent, facilities, equipment or processes for this procedure or affect any planning standard elements. This change does not affect the PNPS Emergency Plan. No further evaluation is required for this change.

Items 6 & 7: These changes provide added direction to ensure that no business network computers, monitors and printers are overlooked during the equipment walkdown. These changes do not change intent, facilities, equipment or processes for this procedure or affect any planning standard elements. These changes do not affect the PNPS Emergency Plan. No further evaluation is required for these changes.

Item 8: This change ensures that the Disaster Recovery Computer and printer are available for use. This computer is not connected to the business network, so it is being singled out as a separate section. This change does not change intent, facilities, equipment or processes for this procedure or affect any planning standard elements. This change does not affect the PNPS Emergency Plan. No further evaluation is required for this change.

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Part VI. Signatures:		
Preparer Name (Print) James R. Parmenter Sr. Emergency Planner	Preparer Signature James R Parmenter	Date: 2/2/16
(Optional) Reviewer Name (Print)	Reviewer Signature	Date:
Reviewer Name (Print) Duane White Nuclear EP Project Manager	Reviewer Signature Duane White	Date: 2-2-2016
Approver Name (Print) DM Calabrese EP manager or designee	Approver Signature DM Calabrese	Date: 2/4/2016