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NLS2016025
April 27, 2016

50.54(q)

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Emergency Plan Implementing Procedures
Cooper Nuclear Station, Docket No. 50-298, DPR-46

Dear Sir or Madam:

The purpose of this letter is to report a change to the following Emergency Plan Implementing Procedures (EPIP) and provide a summary of the associated 10 CFR 50.54(q) analyses for the changes to the EPIPs:

EPIP 5.7.1	Revision 54	Emergency Classification
EPIP 5.7.8	Revision 27	Activation of OSC
EPIP 5.7.16	Revision 25	Release Rate Determination
EPIP 5.7.17.1	Revision 0	Dose Assessment (Manual)
EPIP 5.7.17	Revision 45	CNS-DOSE Assessment
EPIP 5.7.20	Revision 27	Protective Action Recommendations

This letter contains no commitments.

If you have any questions regarding this submittal, please contact me at (402) 825-2788.

Sincerely,

Jim Shaw

Jim Shaw
Licensing Manager

/bk

A045
NRR

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Attachments:
1. Report of Change and Summary of 50.54(q) Analyses - Emergency Plan Implementing Procedure 5.7.1, Revision 54
2. Report of Change and Summary of 50.54(q) Analysis - Emergency Plan Implementing Procedure 5.7.8, Revision 27
3. Report of Change and Summary of 50.54(q) Analyses - Emergency Plan Implementing Procedure 5.7.16, Revision 25
4. Report of Change and Summary of 50.54(q) Analysis - Emergency Plan Implementing Procedure 5.7.17.1, Revision 0
5. Report of Change and Summary of 50.54(q) Analyses - Emergency Plan Implementing Procedure 5.7.17, Revision 45
6. Report of Change and Summary of 50.54(q) Analysis - Emergency Plan Implementing Procedure 5.7.20, Revision 27

Enclosures:
1. Emergency Plan Implementing Procedure 5.7.1, Revision 54
2. Emergency Plan Implementing Procedure 5.7.8, Revision 27
3. Emergency Plan Implementing Procedure 5.7.16, Revision 25
4. Emergency Plan Implementing Procedure 5.7.17.1, Revision 0
5. Emergency Plan Implementing Procedure 5.7.17, Revision 45
6. Emergency Plan Implementing Procedure 5.7.20, Revision 27

cc: Regional Administrator, w/ attachments and enclosures (2)
USNRC – Region IV

Director, Spent Fuel Project Office, w/ attachments and enclosures
Office of Nuclear Material Safety and Safeguards

Senior Resident Inspector, w/ attachments (enclosures per controlled document distribution)
USNRC – CNS

NPG Distribution, w/ attachments and w/o enclosures

CNS Records, w/ attachments and w/o enclosures

NEI 99-01 Basis:

This EAL serves as precursor to a loss of heat removal. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV level decrease and potential core uncovery. This condition will result in a minimum classification of Alert. The low-low ECCS Actuation setpoint was chosen because it is a recognized setpoint. The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

In Cold Shutdown Mode, the RCS will normally be intact and standard RPV level monitoring means are available. In the Refueling Mode, the RCS is not intact and RPV level may be monitored by different means, including the ability to monitor level visually.

In the Refueling Mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RPV inventory event, the Operators would need to determine that RPV inventory loss was occurring by observing sump level changes listed in Table C-1. Sump 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. The 15 minute duration for the loss of level indication was chosen because it is half of the CS2.1 Site Area Emergency EAL duration. The 15 minute duration allows CA2.1 to be an effective precursor to CS2.1. Significant fuel damage is not expected to occur until the core has been uncovered for > 1 hour. Therefore, this EAL meets the definition for an Alert.

If RPV level continues to decrease, then escalation to Site Area Emergency will be via EAL CS2.1.

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CNS Basis:

The threshold RPV level of -42 in. is the low-low ECCS actuation setpoint (Reference 5).

RPV level is normally monitored using the following instruments (Reference 3, 4):

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.).
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.).
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.).
- Narrow Range RFC-LI-94A, B & C (0 to 60 in.).
- Shutdown Range NBI-LI-86 (0 to 400 in.).

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV (Reference 1). 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 (Reference 2). If the make-up 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.

CNS Basis Reference(s):

1. System Operating Procedure 2.2.27, Equipment, Floor, and Chemical Drain System.
2. System Operating Procedure 2.2.69, Residual Heat Removal System.
3. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
4. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
5. Technical Specification Table 3.3.5.1-1.

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 (NOTE 4), RPV level < -48 in.

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 requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

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 a 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 or AG1.1.

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CNS Basis:

When RPV level decreases to -48 in., water level is 6 in. below the low-low ECCS actuation setpoint: -42 in. - 6 in. = -48 in. (Reference 4).

RPV level is normally monitored using the following instruments (Reference 2, 3):

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.).
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.).
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.).
- Narrow Range RFC-LI-94A, B & C (0 to 60 in.).
- Shutdown Range NBI-LI-86 (0 to 400 in.).

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

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 Primary Containment or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety (Reference 1).

CNS Basis Reference(s):

1. Administrative Procedure 0.50.5, Outage Shutdown Safety.
2. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
3. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
4. Technical Specification Table 3.3.5.1-1.

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 2 - RPV Level

Initiating Condition: Loss of RPV inventory affecting core decay heat removal capability

EAL:

CS2.2 Site Area Emergency

With Containment Closure established (NOTE 4), RPV level < -158 in.

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 requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

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 a 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 CG2.1or AG1.1.

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CNS Basis:

When RPV level drops to -158 in., core uncovery is about to occur (Reference 4).

RPV level is normally monitored using the following instruments (Reference 2, 3):

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.).
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.).
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.).
- Narrow Range RFC-LI-94A, B & C (0 to 60 in.).
- Shutdown Range NBI-LI-86 (0 to 400 in.).

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

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 Primary Containment or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety (Reference 1).

CNS Basis Reference(s):

1. Administrative Procedure 0.50.5, Outage Shutdown Safety.
2. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
3. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
4. NEDC 97-089.

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 • Suppression pool water level rise • RPV make-up rate rise • Observation of unisolable RCS leakage

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Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

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 a 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 CG2.2 or AG1.1.

In the Cold Shutdown Mode, normal RPV Level Instrumentation Systems will usually be available. In the Refueling Mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will usually 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 level changes listed in Table C-1. Sump 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.

The 30 minute duration allows sufficient time for actions to be performed to recover inventory control equipment.

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

CNS Basis:

RPV level is normally monitored using the instruments in Figure C-3 (Reference 3, 5).

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.).
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.).
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.).
- Narrow Range RFC-LI-94A, B & C (0 to 60 in.).
- Shutdown Range NBI-LI-86 (0 to 400 in.).

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Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

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 or erratic Source Range Monitor (SRM) indication:

- 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 (Reference 1). 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 water level could be indicative of RHR valve misalignment or leakage (Reference 2). If the make-up 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.
- Source Range Monitor (SRM) indication is provided in the Main Control Room by NMS-I-43A-D, SRM A-D LOG COUNT RATE, NM-NR-45, SRM 2 PEN RECORDER, and SPDS (Reference 4). Erratic Source Range Monitor indication specified is caused by increased boiling in the vicinity of the SRMs as water level lowers and exposed fuel heats up. This instability will be greatly magnified when compared with normal source range monitor behavior. Additionally, once the water level falls below the SRM level, lack of thermalized neutron populations in the area will cause the SRMs to read abnormally low.

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At some plants, containment high range area radiation monitors are specified as an alternate means of detecting loss of water shielding above the core and possible core uncovery. The CNS Primary Containment radiation monitors are designated RA-RM-40A and B. The Containment High Range Monitoring System monitors gamma radiation levels of 1 R/hr to 10^7 R/hr in the reactor drywell area. The system consists of two ion chambers in the drywell and two readout modules in Panels 9-2 and 9-10. The detectors (RA-RE-40A and B) are located 180° from each other in the drywell above elevation 901' 6". The elevation of the bottom of the RPV bottom head is 917'. The elevation of the top of active fuel in the Primary Containment is approximately 947'. The detectors are approximately 45' 6" below the top of active fuel. RCS piping, components, and drywell structural members are positioned between the detectors and the reactor core. Due to the relative location of these detectors with respect to the top of active fuel, the CNS containment high range radiation monitors cannot be utilized for detection of loss of RPV inventory above the core. Additionally, no other installed Radiation Monitoring System exists that can be utilized for the function.

CNS Basis Reference(s):

1. System Operating Procedure 2.2.27, Equipment, Floor, and Chemical Drain System.
2. System Operating Procedure 2.2.69, Residual Heat Removal System.
3. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
4. Instrument Operating Procedure 4.1.1, Source Range Monitoring System.
5. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.

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 < -158 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 ≥ 6% H_2 in drywell or torus **AND** ≥ 5% O_2 in drywell or torus • Unplanned rise in PC pressure • Secondary Containment area radiation > 1000 mR/hr (EOP-5A Table 10)

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 requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.

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Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

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.

Analysis indicates that core damage may occur within an hour following continued core uncovery; therefore, conservatively, 30 minutes was chosen.

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

Containment Closure is the action taken to secure containment (Primary or Secondary) and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure should not be confused with refueling containment integrity as defined in Technical Specifications. Site shutdown contingency plans typically provide for re-establishing Containment Closure following a loss of heat removal or RPV inventory functions. If the closure is re-established prior to exceeding the temperature or level thresholds of the RCS barrier and Fuel Clad barrier EALs, escalation to General Emergency would not occur. If Containment Closure is re-established prior to exceeding the 30 minute core uncovery time limit, then escalation to GE would not occur.

The use of Secondary Containment radiation monitors should provide indication of increased release that may be indicative of a challenge to Secondary Containment. The radiation monitor values are based on the EOP "maximum safe values" because these values are easily recognizable and have an emergency basis.

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In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gasses in Primary Containment. However, Primary Containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists.

CNS Basis:

When RPV level drops to -158 in., core uncovery is about to occur (Reference 7).

RPV level is normally monitored using the following instruments (Reference 3, 4):

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.).
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.).
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.).
- Narrow Range RFC-LI-94A, B & C (0 to 60 in.).
- Shutdown Range NBI-LI-86 (0 to 400 in.).

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

Four conditions are associated with a challenge to Primary Containment (PC) integrity:

1. Containment Closure is the action taken to secure Primary Containment or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety (Reference 1).

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2. 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. The specified values for this threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen) (Reference 5).

3. 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 indicates Containment Closure cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.

4. 1,000 mR/hr is the Secondary Containment Maximum Safe Operating radiation value. Exceeding this value is 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-5A, Secondary Containment Control, Table 10. As indicated by NOTE 5 in EOP-5A, Table 10, RP Surveys and ARM Teledosimetry System may be used for these indications (Reference 6).

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EOP-5A Table 10 - Secondary Containment Radiation Levels

10 SECONDARY CONTAINMENT RADIATION LEVELS (5)
SPDS 15

Maximum Normal Operating Value			Maximum Safe Operating Value		
Area	Any ARM Alarmed	Range (mR/hr)	Area	Value (mR/hr)	Actual Value
FUEL POOL AREA	RMA-RA-1	100 - 10^6	1001' El.	1000	
FUEL POOL AREA	RMA-RA-2	.01 - 100	1001' El.		
RWCU PRECOAT AREA	RMA-RA-4	0.1 - 1000	958' El.		
RWCU SLUDGE AND DECANT PUMP AREA	RMA-RA-5	0.1 - 1000	931' El.	1000	
CRD HYDRAULIC EQUIP AREA (SOUTH)	RMA-RA-8	.01 - 100	903' El.		
CRD HYDRAULIC EQUIP AREA (NORTH)	RMA-RA-9	.01 - 100			
HPCI PUMP ROOM	RMA-RA-10	.01 - 100	HPCI Room		
RHR PUMP ROOM, (SOUTHWEST)	RMA-RA-11	.01 - 100	SW Quad	1000	
TORUS HPV AREA (SOUTHWEST)	RMA-RA-27	1.0 - 10000	SW Torus		
RHR PUMP ROOM, (NORTHWEST)	RMA-RA-12	.01 - 100	NW Quad	1000	
RCIC/CORE SPRAY PUMP ROOM, (NORTHEAST)	RMA-RA-13	.01 - 100	NE Quad	1000	
CORE SPRAY PUMP ROOM, (SOUTHEAST)	RMA-RA-14	.01 - 100	SE Quad	1000	

NOTE 5

Area radiation levels can be monitored by RP surveys or ARM teledosimetry system

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CNS Basis Reference(s):

1. Administrative Procedure 0.50.5, Outage Shutdown Safety.
2. System Operating Procedure 2.2.60.1, Containment H_2/O_2 Monitoring System.
3. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
4. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
5. AMP-TBD00, Step PC/H.
6. EOP-5A, Secondary Containment Control, Table 10.
7. NEDC 97-089.

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 core uncovery 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.

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 • Suppression pool water level rise • RPV make-up rate rise • Observation of unisolable RCS leakage

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Table C-5 Containment Challenge Indications
• Containment Closure **not** established (Note 4) • Deflagration concentrations exist inside PC $\geq$ 6% H_2 in drywell or torus **AND** $\geq$ 5% O_2 in drywell or torus • Unplanned rise in PC pressure • Secondary Containment area radiation > 1000 mR/hr (EOP-5A Table 10)

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 requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

NEI 99-01 Basis:

In the Cold Shutdown Mode, normal RPV level and RPV Level Instrumentation Systems will normally be available. 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 level changes. Sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Primary Containment to ensure they are indicative of RCS leakage.

In the Refueling Mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RPV inventory event, the Operators would need to determine that RPV inventory loss was occurring by observing sump level changes listed in Table C-1.

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For both Cold Shutdown and Refueling Modes, sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Primary Containment to ensure they are indicative of RCS leakage.

A number of variables, such as initial RPV level or shutdown heat removal system design, can have a significant impact on heat removal capability challenging the fuel clad barrier. Analysis in the above references indicates that significant core damage may occur within an hour following continued core uncovery; therefore, conservatively, 30 minutes was chosen.

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

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

Containment Closure is the action taken to secure either Primary Containment or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure should not be confused with refueling containment integrity as defined in Technical Specifications. Site shutdown contingency plans provide for re-establishing Containment Closure following a loss of heat removal or RPV inventory functions. If the closure is re-established prior to exceeding the temperature or level thresholds of the RCS Barrier and Fuel Clad Barrier EALs, escalation to General Emergency would not occur. If Containment Closure is re-established prior to exceeding the 30 minute core uncovery time limit, then escalation to GE would not occur.

The use of Secondary Containment radiation monitors should provide indication of increased release that may be indicative of a challenge to Secondary Containment. The radiation monitor values are based on the EOP "maximum safe values" because these values are easily recognizable and have an emergency basis.

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In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gasses in Primary Containment. However, Primary Containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists.

CNS Basis:

RPV level is normally monitored using the following instruments (Reference 5, 7):

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.).
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.).
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.).
- Narrow Range RFC-LI-94A, B & C (0 to 60 in.).
- Shutdown Range NBI-LI-86 (0 to 400 in.).

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

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 or erratic Source Range Monitor (SRM) indication:

- 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 (Reference 2). 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 water level could be indicative of RHR valve misalignment or leakage (Reference 4). If the make-up 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.

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- Source Range Monitor (SRM) indication is provided in the Main Control Room by NMS-I-43A-D, SRM A-D LOG COUNT RATE, NM-NR-45, SRM 2 PEN RECORDER, and SPDS (Reference 6). Erratic Source Range Monitor indication specified is caused by increased boiling in the vicinity of the SRMs as water level lowers and exposed fuel heats up. This instability will be greatly magnified when compared with normal source range monitor behavior. Additionally, once the water level falls below the SRM level, lack of thermalized neutron populations in the area will cause the SRMs to read abnormally low.

At some plants, containment high range area radiation monitors are specified as an alternate means of detecting loss of water shielding above the core and possible core uncovery. The CNS Primary Containment radiation monitors are designated RA-RM-40A and RA-RM-40B. The Containment High Range Monitoring System monitors gamma radiation levels of 1 R/hr to 10^7 R/hr in the reactor drywell area. The system consists of two ion chambers in the drywell and two readout modules in Panels 9-2 and 9-10. The detectors (RA-RE-40A and RA-RE-40B) are located 180° from each other in the drywell above elevation 901' 6". The elevation of the bottom of the RPV bottom head is 917'. The elevation of the top of active fuel in the Primary Containment is approximately 947'. The detectors are approximately 45' 6" below the top of active fuel. RCS piping, components, and drywell structural members are positioned between the detectors and the reactor core. Due to the relative location of these detectors, with respect to the top of active fuel, the CNS containment high range radiation monitors cannot be utilized for detection of loss of RPV inventory above the core. Additionally, no other installed Radiation Monitoring System exists that can be utilized for the function.

Four conditions are associated with a challenge to Primary Containment (PC) integrity:

1. Containment Closure is the action taken to secure Primary Containment or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety (Reference 1).

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2. 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. The specified values for this threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen) (Reference 8).

3. 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 indicates Containment Closure cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.

4. 1,000 mR/hr is the Secondary Containment Maximum Safe Operating radiation value. Exceeding this value is 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-5A, Secondary Containment Control, Table 10. As indicated by NOTE 5 in EOP-5A, Table 10, RP Surveys and ARM Teledosimetry System may be used for these indications (Reference 9).

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EOP-5A Table 10 - Secondary Containment Radiation Levels

10 SECONDARY CONTAINMENT RADIATION LEVELS 5
SPDS 15

Maximum Normal Operating Value			Maximum Safe Operating Value		
Area	Any ARM Alarmed	Range (mR/hr)	Area	Value (mR/hr)	Actual Value
FUEL POOL AREA	RMA-RA-1	$100 - 10^6$	1001' El.	1000	
FUEL POOL AREA	RMA-RA-2	.01 - 100	1001' El.		
RWCU PRECOAT AREA	RMA-RA-4	0.1 - 1000	958' El.		
RWCU SLUDGE AND DECANT PUMP AREA	RMA-RA-5	0.1 - 1000	931' El.	1000	
CRD HYDRAULIC EQUIP AREA (SOUTH)	RMA-RA-8	.01 - 100	903' El.		
CRD HYDRAULIC EQUIP AREA (NORTH)	RMA-RA-9	.01 - 100			
HPCI PUMP ROOM	RMA-RA-10	.01 - 100	HPCI Room		
RHR PUMP ROOM, (SOUTHWEST)	RMA-RA-11	.01 - 100	SW Quad	1000	
TORUS HPV AREA (SOUTHWEST)	RMA-RA-27	1.0 - 10000	SW Torus		
RHR PUMP ROOM, (NORTHWEST)	RMA-RA-12	.01 - 100	NW Quad	1000	
RCIC/CORE SPRAY PUMP ROOM, (NORTHEAST)	RMA-RA-13	.01 - 100	NE Quad	1000	
CORE SPRAY PUMP ROOM, (SOUTHEAST)	RMA-RA-14	.01 - 100	SE Quad	1000	

NOTE 5

Area radiation levels can be monitored by RP surveys or ARM teledosimetry system

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CNS Basis Reference(s):

1. Administrative Procedure 0.50.5, Outage Shutdown Safety.
2. System Operating Procedure 2.2.27, Equipment, Floor, and Chemical Drain System.
3. System Operating Procedure 2.2.60.1, Containment H_2/O_2 Monitoring System.
4. System Operating Procedure 2.2.69, Residual Heat Removal System.
5. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
6. Instrument Operating Procedure 4.1.1, Source Range Monitoring System.
7. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
8. AMP-TBD00, Step PC/H.
9. EOP-5A, Secondary Containment Control, Table 10.

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

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

NEI 99-01 Basis:

This EAL is an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. 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. In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling Mode. Entry into Cold Shutdown conditions may be attained within hours of operating at power. Entry into the Refueling Mode procedurally may not occur for many hours after the reactor has been shut down. Thus, the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the Refueling Mode with irradiated fuel in the RPV (note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling). In addition, the Operators should be able to monitor RCS temperature and RPV level so that escalation to the Alert level via CA2.1 or CA3.1 will occur if required.

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During refueling operations, the level in the RPV will normally be maintained above the vessel flange. Refueling evolutions that decrease water level below the vessel 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. Normal means of core temperature indication and RPV level indication may not be available in the Refueling Mode. Redundant means of RPV level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. Escalation to the Alert level via CA3.1 may therefore be required should an unplanned event result in RCS temperature exceeding the Technical Specification Cold Shutdown temperature limit for an extended period of time. The allowed time varies and is dependent on the status of the Primary Containment and Secondary Containment barriers and the integrity of the RCS barrier.

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

CNS Basis:

Several instruments are capable of providing indication of RPV temperature with respect to the Technical Specification cold shutdown temperature limit (212°F) (Reference 4). These include (Reference 1, 2):

- NBI-TR-89, REACTOR VESSEL METAL TEMPERATURE RECORDER (Panel 9-21).
- Vessel Drain, PMIS Point M180, or NBI-TR-89 - Point 06 if M180 is not available.
- Vessel Bottom Head, PMIS Point M184, or NBI-TR-89 - Point 10 if M184 is not available.
- Bottom Head Adjacent to Support Skirt, PMIS Point M183, or NBI-TR-89 - Point 09.
- RR-TR-165, RR SUCTION & FEEDWATER TEMP (Panel 9-4).

PMIS Points M174 through M185 can be used to monitor RPV temperatures. Thermocouples associated with computer Points M180, M183, and M185 do not respond as quickly nor register as high a temperature as other thermocouples due to their locations.

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Inservice leak testing, hydrostatic testing, and control rod scram time testing in which RCS temperature is intentionally raised above 212°F per Technical Specification LCO 3.10.1 are not applicable to this EAL (Reference 3).

CNS Basis Reference(s):

1. General Operating Procedure 2.1.1, Startup Procedure.
2. System Operating Procedure 2.2.69.2, RHR System Shutdown Operations.
3. Technical Specifications LCO 3.10.1.
4. Technical Specifications Table 1.1-1.

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 - Refueling

NEI 99-01 Basis:

This EAL is included as an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. 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. In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling Mode. Entry into Cold Shutdown conditions may be attained within hours of operating at power. Entry into the Refueling Mode procedurally may not occur for many hours after the reactor has been shut down. Thus, the heatup threat and therefore the threat to damaging the fuel clad may be lower for events that occur in the Refueling Mode with irradiated fuel in the RPV. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling outage. In addition, the Operators should be able to monitor RCS temperature and RPV level so that escalation to the Alert level via CA2.1 or CA3.1 will occur if required.

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During refueling operations, the level in the RPV will normally be maintained above the vessel flange. Refueling operations that lower water level below the vessel 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 the Alert level via CA3.1 may therefore be required should an unplanned event result in RCS temperature exceeding the Technical Specification Cold Shutdown temperature limit for an extended period of time. The allowed time varies and is dependent on the status of the Primary Containment and Secondary Containment barriers and the integrity of the RCS barrier.

Unlike the Cold Shutdown Mode, normal means of core temperature indication and RCS level indication may not be available in the Refueling Mode. Redundant means of RPV level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the Cold Shutdown or Refueling Modes, this EAL would result in declaration of an Unusual Event if either temperature or level indication cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert would be under CA2.1 based on an inventory loss or CA3.1 based on exceeding its temperature criteria (212°F, Reference 3).

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

CNS Basis:

RPV level is normally monitored using the following instruments (Reference 3, 4):

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.).
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.).
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.).
- Narrow Range RFC-LI-94A, B & C (0 to 60 in.).
- Shutdown Range NBI-LI-86 (0 to 400 in.).

RPV level monitoring also includes the ability to monitor level visually in Refueling Mode consistent with escalation EAL CA2.1.

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Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

Several instruments are capable of providing indication of RPV temperature with respect to the Technical Specification cold shutdown temperature limit (212°F). These include (Reference 1, 2):

- NBI-TR-89, REACTOR VESSEL METAL TEMPERATURE RECORDER (Panel 9-21).
- Vessel Drain, PMIS Point M180, or NBI-TR-89 - Point 06 if M180 is not available.
- Vessel Bottom Head, PMIS Point M184, or NBI-TR-89 - Point 10 if M184 is not available.
- Bottom Head Adjacent to Support Skirt, PMIS Point M183, or NBI-TR-89 - Point 09.
- RR-TR-165, RR SUCTION & FEEDWATER TEMP (Panel 9-4).

PMIS Points M174 through M185 can be used to monitor RPV temperatures. Thermocouples associated with computer Points M180, M183, and M185 do not respond as quickly nor register as high a temperature as other thermocouples due to their locations.

CNS Basis Reference(s):

1. General Operating Procedure 2.1.1, Startup Procedure.
2. System Operating Procedure 2.2.69.2, RHR System Shutdown Operations.
3. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
4. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.

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 **EITHER**:

RCS temperature > 212°F for > Table C-3 duration (NOTE 4)

OR

RPV pressure increase > 10 psig due to a loss of RCS cooling

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 requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.

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.

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Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

NEI 99-01 Basis:

The first condition of this EAL addresses events in which RCS temperature exceeds the CU3.1 EAL threshold of 212°F (Reference 6) for the durations identified in Table C-3.

Table C-3, Duration #3, addresses complete loss of functions required for core cooling during Refueling and Cold Shutdown Modes when neither Containment Closure nor RCS integrity are established. RCS integrity is in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown Mode of operation. No delay time is allowed for Duration #3 because the evaporated reactor coolant that may be released into the containment during this heatup condition could also be directly released to the environment.

Table C-3, Duration #2, addresses the complete loss of functions required for core cooling for > 20 minutes during Refueling and Cold Shutdown Modes when Containment Closure is established but RCS integrity is not established. RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown Mode of operation. The allowed 20 minute time frame was included to allow Operator action to restore the heat removal function, if possible. The allowed time frame is consistent with the guidance provided by Generic Letter 88-17, Loss of Decay Heat Removal, and is believed to be conservative given that a low pressure containment barrier to fission product release is established. The table note indicates that this duration is not applicable if actions are successful in restoring a RCS Heat Removal System to operation and RCS temperature is being reduced within the 20 minute time frame.

Table C-3, Duration #1, addresses complete loss of functions required for core cooling for > 60 minutes during Refueling and Cold Shutdown Modes when RCS integrity is established. As in Durations #2 and #3, RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown Mode of operation. The status of Containment Closure in this EAL is immaterial given that the RCS is providing a high pressure barrier to fission product release to the environment. The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety.

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The 10 psig pressure increase covers situations where, due to high decay heat loads, the time provided to restore temperature control should be < 60 minutes. The table note indicates that Duration #1 is not applicable if actions are successful in restoring a RCS Heat Removal System to operation and RCS temperature is being reduced within the 60 minute time frame assuming that the RCS pressure increase has remained < 10 psig.

Escalation to Site Area would be via CS2.1 should boiling result in significant RPV level loss leading to core uncovery.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary unplanned excursion above 212°F when the heat removal function is available and either the RCS is intact or Containment Closure is established.

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.

CNS Basis:

A 10 psig RPV pressure increase can be read on (Reference 3, 4):

- RFC-PI-90A (Panel 9-5, 0 - 1200 psig).
- RFC-PI-90B (Panel 9-5, 0 - 1200 psig).
- RFC-PI-90C (Panel 9-5, 0 - 1200 psig).
- Reactor pressure on PMIS Point B025.

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Several instruments are capable of providing indication of RPV temperature with respect to the Technical Specification cold shutdown temperature limit (212°F). These include (Reference 1, 2):

- NBI-TR-89, REACTOR VESSEL METAL TEMPERATURE RECORDER (Panel 9-21).
- Vessel Drain, PMIS Point M180, or NBI-TR-89 - Point 06 if M180 is not available.
- Vessel Bottom Head, PMIS Point M184, or NBI-TR-89 - Point 10 if M184 is not available.
- Bottom Head Adjacent to Support Skirt, PMIS Point M183, or NBI-TR-89 - Point 09.
- RR-TR-165, RR SUCTION & FEEDWATER TEMP (Panel 9-4).

PMIS Points M174 through M185 can be used to monitor RPV temperatures. Thermocouples associated with computer Points M180, M183, and M185 do not respond as quickly nor register as high a temperature as other thermocouples due to their locations.

Inservice leak testing, hydrostatic testing, and control rod scram time testing in which RCS temperature is intentionally raised above 212°F per Technical Specification LCO 3.10.1 are not applicable to this EAL (Reference 5).

CNS Basis Reference(s):

1. General Operating Procedure 2.1.1, Startup Procedure.
2. System Operating Procedure 2.2.69.2, RHR System Shutdown Operations.
3. Instrument Operating Procedure 4.6.2, Reactor Vessel Pressure Indication.
4. Emergency Procedure 5.9SAMG, Severe Accident Management Guidance, Technical Support Guidelines attachment (CPA TSG).
5. Technical Specifications LCO 3.10.1.
6. Technical Specifications Table 1.1-1.

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 4 - Communications

Initiating Condition: Loss of **all** on-site or off-site communications capabilities

EAL:

CU4.1 Unusual Event

Loss of **all** Table C-2 on-site (internal) communication methods affecting the ability to perform routine operations

OR

Loss of **all** Table C-2 off-site (external) communication methods affecting the ability to perform off-site notifications

Table C-2 Communications Systems

System	Onsite (internal)	Offsite (external)
Station Intercom System "Gaitronics"	X	
Site UHF Radio Consoles	X	
Radio Paging System	X	
Alternate Intercom	X	
Sound Power System	X	
CNS On-Site Cell Phone System	X	X
Telephone system (PBX)	X	X
Federal Telecommunications System (FTS 2001)		X
Local Telephones (C.O. Lines)		X
CNS State Notification Telephones		X
Satellite Telephones		X

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, D - Defueled

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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 problems with off-site authorities. The loss of off-site communications ability is expected to be significantly more comprehensive than the condition addressed by 10CFR50.72.

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

The Table C-2 list for on-site communications loss encompasses the loss of all means of routine communications (e.g., commercial telephones, sound powered phone systems, page party system, and radios/walkie talkies).

The Table C-2 list for off-site communications loss encompasses the loss of all means of communications with federal off-site authorities. This should include the ENS, commercial telephone lines, telecopy transmissions, and dedicated phone systems.

CNS Basis:

NOTE – EPIP 5.7COMMUN has more detail on each of the communications systems covered by this EAL.

On-site/off-site communications include one or more of the systems listed in Table C-2 (Reference 1).

- Station Intercom System "Gaitronics": Permits communication between the different parts of the plant and it also incorporates a public address system for plant wide announcements.
- Site UHF Radio Consoles: The site UHF radio system uses four repeaters; Base 1 and Base 2 are used by Operations, Base 3 and Base 4 are used by Security. These repeaters operate on different frequencies. All remote control, portable, and mobile units are capable of selecting either repeater.

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- Radio Paging System: CNS leases pagers and radio paging services from a telecommunications company. Pagers are issued to various Management and Emergency Response personnel at CNS and other NPPD locations. Pagers can be activated from any touch-tone phone, on-site or off-site.
- Alternate Intercom: Provides an alternate in-plant communications network utilizing the back-up tone commander PBX System. This system is located in the ERP shack and has battery back-up.
- Sound Powered System.
- CNS On-Site Cell Phone System.
- Telephone System (PBX): Provides voice communication between virtually all buildings, offices, and operation facilities within the station. The telephone system also provides communications between the plant and off-site facilities via the telephone switchboard network. The system allows Operating Crews to alert plant personnel in emergencies. The telephone company provides the normal and leased line services.
- Federal Telecommunications System (FTS 2001): The Health Physics Network (HPN) and Emergency Notification System (ENS) provides communications between NRC and CNS during an emergency.
- Local Telephones (C.O. Lines).
- CNS State Notification Telephones: The CNS State Notification Telephone System is the primary means for the plant to make emergency notifications to state and local authorities. This system provides direct communication with the Nebraska State Patrol, the Missouri State Patrol, the Atchison County Sheriff's Department, and the Nemaha and Richardson County Sheriff's Departments.
- Satellite Telephones.

This EAL is the cold condition equivalent of the hot condition EAL SU8.1.

CNS Basis Reference(s):

1. EPIP 5.7COMMUN, Communications, Emergency Response Facility Communication Equipment attachment.

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 5 - Inadvertent Criticality

Initiating Condition: Inadvertent criticality

EAL:

CU5.1	Unusual Event
An unplanned sustained positive period observed on nuclear instrumentation	

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

NEI 99-01 Basis:

This EAL addresses criticality events that occur in Cold Shutdown or Refueling Modes (NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States) such as fuel mis-loading events. This EAL indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification. This EAL excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated) which are addressed in the companion EAL SU2.1.

The term "sustained" is used in order to allow exclusion of expected short term positive periods from planned fuel bundle or control rod movements during core alteration. These short term positive periods are the result of the increase in neutron population due to subcritical multiplication.

Escalation to higher classification levels would be by the judgment EALs in Category H (EAL HA6.1, HS6.1, or HG6.1).

CNS Basis:

SRM A-D period Meters NMS-I-44A-D on Panel 9-5 identify this condition as well as Panel 9-5 amber light and SRM Period (< 50 sec.) Annunciator 9-5-1/F-8 (Reference 1, 2). However, a SRM period alarm caused by SRM channel noise does not result in entry into this EAL (Reference 1).

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CNS Basis Reference(s):

1. Alarm Procedure 2.3_9-5-1, Panel 9-5 - Annunciator 9-5-1, F-8, SRM Period.
2. Instrument Operating Procedure 4.1.1, Source Range Monitoring System.

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

< 105 VDC bus voltage indications on **all** Technical Specification required 125 VDC 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 - Refueling

NEI 99-01 Basis:

The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations.

This EAL is intended to be anticipatory in as much as the Operating Crew may not have necessary indication and control of equipment needed to respond to the loss.

The plant will routinely perform maintenance on a train related basis during shutdown periods. The required buses are the minimum allowed by Technical Specifications for the mode of operation. It is intended that the loss of the operating (operable) train is to be considered. If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will be per CA3.1.

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

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CNS Basis:

105 VDC is the minimum design bus voltage (Reference 4).

The 125 VDC System supplies DC power to conventional station emergency equipment and selected Safeguard System loads. 125 VDC Distribution Panels supply control and instrument power for annunciators control logic power and protective relaying. Figure C-2 illustrates the 125 VDC Power System (Reference 3).

If 125 VDC Distribution Panel A is lost, the following major equipment is affected: RRMG A speed and breaker control, 4160V Bus 1A, 1E, and 1F breaker control and undervoltage logics, 480V Bus 1A and 1F breaker control, the right light in all Control Room annunciators, annunciator panels for Water Treatment, RHR A Gland Water, Auxiliary Steam Boiler C, DG-1 starting and breaker control logics, CS A, RCIC, and RHR A control logics, TIP valve control monitors, main generator voltage regulation, RFPT A trip logic, and ARI solenoid valve power.

If 125 VDC Distribution Panel B is lost, the following major equipment is affected: RRMG B speed and breaker control, 4160V Bus 1B and 1G breaker control and undervoltage logics, 480V Bus 1B and 1G breaker control, the left light in all Control Room annunciators, annunciator panels for ALRW, RHR B Gland Water, Auxiliary Steam Boiler D, DG-2 starting and breaker control logics, CS B, HPCI, and RHR B control logics, main generator trip logic, main generator and transformer protective relaying, bypass valves fail to control pressure after turbine trip, and RFPT B trip logic.

Battery chargers receive their power from 460V critical motor control centers. Each 125 VDC bus receives power from either a 125 VDC battery or a 125 VDC battery charger. The battery chargers receive their power from 460V critical motor control centers. The 250 VDC System supplies DC power to conventional station emergency equipment and selected Safeguard System loads. Although 250 VDC Buses 1A and 1B provide vital DC emergency power, 250 VDC Safety System loads (such as motor operated valves) also require 125 VDC control power. Loss of 125 VDC buses alone, therefore, would render most Safeguard System loads inoperable (Reference 4, 5, 6).

This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS7.1.

(continued on next page)

CNS Basis Reference(s):

1. Emergency Procedure 5.3DC125, Loss of 125 VDC.
2. Surveillance Procedure 6.EE.607, 125V Station Battery Modified Performance Discharge Test.
3. BR 3058 DC One Line Diagram.
4. Technical Specifications B 3.8.4.
5. USAR Section VIII-6.2.
6. USAR Section VIII-6.3.

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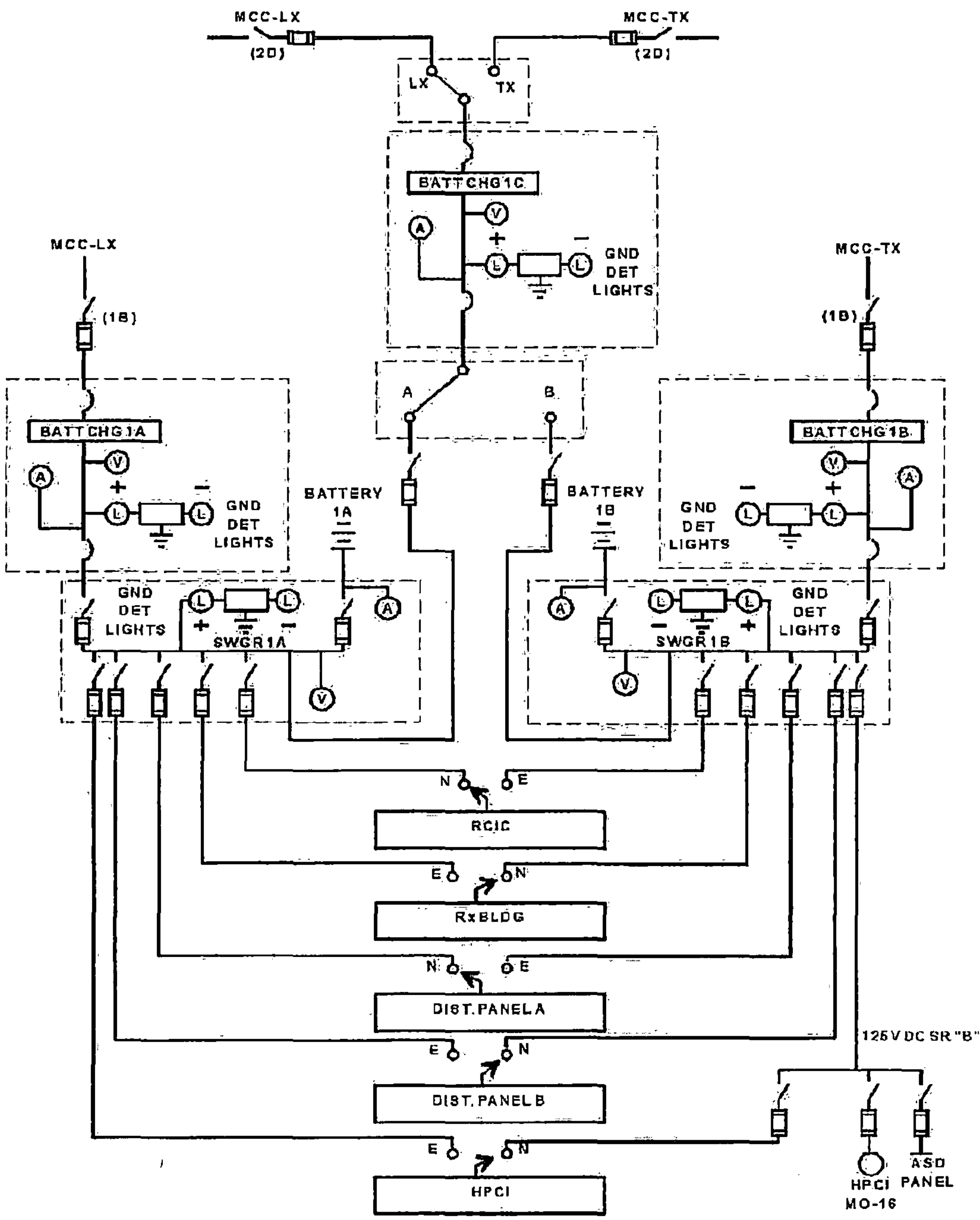
Figure C-2: 125 VDC Power System



Category: H - Hazards And Other Conditions Affecting Plant Safety

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:

- The Seismic Monitor System free field sensor actuated or Alarm B-3/B-1, SEISMIC EVENT
- 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.

As defined in the EPRI-sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989, a "felt earthquake" is: An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of Control Room Operators on-duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated.

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

(continued on next page)

CNS 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 CNS 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.

CNS seismic instrumentation actuates at 0.01 g. The free field seismic sensor, located in the metal enclosure north of the Intake Structure, provides a start signal to the Seismic Monitoring System when ground motion > 0.01 g is sensed. On receipt of this start signal, the Seismic Monitor System indicates or initiates:

- An Event is in-progress.
- An Event has been recorded.
- Annunciator B-3/B-1, SEISMIC EVENT, alarms.
- Recording of the seismic signals from sensors located north of the Intake Structure, in the Reactor Building NW Quad 859', and on the Reactor Building 1001' level.

The National Earthquake Information Center can confirm if an earthquake has occurred in the area of the plant. Refer to the National Earthquake Information Center website:

http://earthquake.usgs.gov/eqcenter/

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

In the absence of operable seismic switches, the Earthquake Magnitude Correlation below may be used to perform an assessment of the relative magnitude of an earthquake.

(continued on next page)

Modified Mericalli Intensity Scale	Description of Effects*	Maximum Acceleration (g)	Richter Magnitude (Descriptor)
I.	Not felt except by a very few under especially favorable conditions.	0.001 to 0.002	1.0 to 2.9 (Very Minor)
II.	Felt only by a few persons at rest, especially on upper floors of buildings.		
III.	Felt quite noticeably by persons indoors, especially on upper floors of buildings. Many people do not recognize it as an earthquake. Standing motor cars may rock slightly. Vibrations similar to the passing of a truck. Duration estimated.	0.003 to 0.010	3.0 to 3.9 (Minor)
IV.	Felt indoors by many, outdoors by few during the day. At night, some awakened. Dishes, windows, doors disturbed; walls make cracking sound. Sensation like heavy truck striking building. Standing motor cars rocked noticeably.		
V.	Felt by nearly everyone; many awakened. Some dishes, windows broken. Unstable objects overturned. Pendulum clocks may stop.	0.011 to 0.029	4.0 to 4.9 (Light)
VI.	Felt by all, many frightened. Some heavy furniture moved; a few instances of fallen plaster. Damage slight.	0.03 to 0.2 (CNS Design Basis OBE = 0.1; CNS Design Basis SSE = 0.2)	5.0 to 5.9 (Moderate)
VII.	Damage negligible in buildings of good design and construction; slight to moderate in well-built ordinary structures; considerable damage in poorly built or badly designed structures; some chimneys broken.		
VIII.	Damage slight in specially designed structures; considerable damage in ordinary substantial buildings with partial collapse. Damage great in poorly built structures. Fall of chimneys, factory stacks, columns, monuments, walls. Heavy furniture overturned.	0.2 to 0.4	6.0 to 6.9 (Strong)
IX.	Damage considerable in specially designed structures; well-designed frame structures thrown out of plumb. Damage great in substantial buildings, with partial collapse. Buildings shifted off foundations.	0.4 to 0.5	7.0 to 7.9 (Major)
X.	Some well-built wooden structures destroyed; most masonry and frame structures destroyed with foundations. Rails bent.		
XI.	Few, if any (masonry), structures remain standing. Bridges destroyed. Rails bent greatly.	> 0.5	8.0 and higher (Great)
XII.	Damage total. Lines of sight and level are distorted. Objects thrown into the air.		

* The Richter magnitude and maximum acceleration (g) in the table above is provided for general comparison and description. Actual seismic intensity and effects will vary.

(continued on next page)

CNS Basis Reference(s):

1. Alarm Procedure 2.3_B-3, Panel B - Annunciator B-3/B-1.
2. Alarm Procedure 2.3_B-3, Panel B - Annunciator B-3/A-1.
3. Instrument Operating Procedure 4.12, Seismic Instrumentation.
4. Emergency Procedure 5.1Quake, Earthquake.
5. USAR Section II-5.2.4 and Table II-5-1.

Category: H - Hazards And Other Conditions Affecting Plant Safety

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 ≥ 100 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 HA1.2.

CNS 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.

The design wind pressure for the station and structures is 30 lb/ft^2 which is equivalent of sustained winds up to 100 mph (Reference 3).

The Protected Area refers to the designated security area around the process buildings and is depicted in Technical Specifications Figure 4.1-1 (Reference 2).

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Sustained winds are of a prolonged duration and, therefore, do not include gusts. Sustained winds are not intermittent or of a transitory nature. Since the inauguration of the Automatic Surface Observation System (ASOS), the National Weather Service has adopted a 2 minute average standard for its sustained wind definition (Reference 1).

CNS Basis Reference(s):

1. National Weather Service webpage "http://www.aoml.noaa.gov/hrd/tcfaq/D4.html".

2. Technical Specifications Figure 4.1-1, Site and Exclusion Area Boundaries and Low Population Zone.

3. USAR Section II-3.2.2.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU1.3 Unusual Event Main 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 HU2.1 and HU3.1.

This EAL is consistent with the definition of an Unusual Event while maintaining the anticipatory nature desired and recognizing the risk to non-safety-related equipment.

Escalation of this emergency classification level, if appropriate, would be to HA1.3 based on damage done by projectiles generated by the failure or by any radiological releases. These latter events would be classified by the Category A, Abnormal Rad Release/Rad Effluent EALs, or the Category F, Fission Product Barrier EALs.

(continued on next page)

CNS Basis:

The main 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.

In addition to potential missile generation, the failure of the rotational elements will cause imbalances in these elements that may be of sufficient magnitude to damage turbine and generator seals resulting in the release of large quantities of lube oil and/or hydrogen.

CNS Basis Reference(s):

None

Category: H - Hazards And Other Conditions Affecting Plant Safety

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-1 area that has the potential to affect safety-related equipment required by Technical Specifications for the current operating mode

Table H-1 Safe Shutdown Areas
• Reactor Building • Control Building • Service Water Pump Room • Diesel Generator Building • Cable Expansion Room

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 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 visible damage via HA1.4 or by other plant conditions.

(continued on next page)

CNS Basis:

The internal flooding areas of concern are listed in Table H-1 (Reference 1-5).

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.

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.

The accumulation of water resulting in a rising water level in the area constitutes flooding.

The Critical Switchgear Rooms are a part of the Reactor Building.

CNS Basis Reference(s):

1. Site Services Procedure 1.1, Station Security.
2. CNS-FP-60, Fire Area Boundary Drawing Index.
3. Drawing CNS-EE-187, CNS Safe Shutdown Component Locations and Emergency Route Lighting - Site Plan.
4. Fire Safety Analysis Calculations.
5. USAR Section XII-2.1.2.1, Principal Class I Structures Required for Safe Shutdown.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU1.5 Unusual Event
High river/forebay water level > 899' MSL
OR
Low river level/forebay < 870' MSL

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.

CNS Basis:

This EAL covers high river/forebay water level conditions that could be a precursor of more serious events as well as low river/forebay water level conditions which may threaten operability of plant cooling systems.

Procedure 5.1FLOOD entry is based on any of the following:

- River water level > 895' MSL.
- Upstream dam failure.
- Projected river water level ≥ 902' MSL within next 36 hours.

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899' MSL is the level associated with the maximum flood of record (Reference 2).

870' MSL is the Minimum Probable river level (Reference 3). A further level drop may threaten availability of cooling systems and heat sink.

The forebay refers to the area between the east Intake Structure wall and the guide wall (Reference 4).

CNS Basis Reference(s):

1. Emergency Procedure 5.1FLOOD, Flood.
2. USAR Section II-4.2.2.2.
3. USAR Section II-4.2.3.1.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting Vital Areas

EAL:

HA1.1 Alert
Seismic event > 0.1 g as indicated by the Seismic Monitor System free field sensor or Alarm B-3/A-1, EMERGENCY SEISMIC HIGH LEVEL
AND
Earthquake confirmed by **any** of the following:
• Earthquake felt in plant
• National Earthquake Information Center
• Control Room indication of degraded performance of systems required for the safe shutdown of the plant

Mode Applicability:

All

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 indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occur on the basis of other EALs (e.g., System Malfunction).

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.

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

CNS Basis:

CNS seismic instrumentation actuates at 0.01 g. The free field seismic sensor, located in the metal enclosure north of the Intake Structure, provides a start signal to the Seismic Monitoring System when ground motion > 0.01 g is sensed. On receipt of this start signal, the Seismic Monitor System indicates or initiates:

- An Event is in-progress.
- An Event has been recorded.
- Annunciator B-3/B-1, SEISMIC EVENT, alarms.
- Recording of the seismic signals from sensors north of the Intake Structure, in the Reactor Building NW Quad 859', and on the Reactor Building 1001' level are recorded.
- Alarm B-3/A-1, EMERGENCY SEISMIC HIGH LEVEL, is received if the seismic activity exceeds 0.1 g.

The National Earthquake Information Center can confirm if an earthquake has occurred in the area of the plant. Refer to the National Earthquake Information Center website:

http://earthquake.usgs.gov/eqcenter/

In the absence of operable seismic switches, the Earthquake Magnitude Correlation below may be used to perform an assessment of the relative magnitude of an earthquake. An earthquake in excess of the CNS Operating Basis Earthquake (OBE) levels should be classified under this EAL.

(continued on next page)

Modified Mericalli Intensity Scale	Description of Effects*	Maximum Acceleration (g)	Richter Magnitude (Descriptor)
I.	Not felt except by a very few under especially favorable conditions.	0.001 to 0.002	1.0 to 2.9 (Very Minor)
II.	Felt only by a few persons at rest, especially on upper floors of buildings.		
III.	Felt quite noticeably by persons indoors, especially on upper floors of buildings. Many people do not recognize it as an earthquake. Standing motor cars may rock slightly. Vibrations similar to the passing of a truck. Duration estimated.	0.003 to 0.010	3.0 to 3.9 (Minor)
IV.	Felt indoors by many, outdoors by few during the day. At night, some awakened. Dishes, windows, doors disturbed; walls make cracking sound. Sensation like heavy truck striking building. Standing motor cars rocked noticeably.		
V.	Felt by nearly everyone; many awakened. Some dishes, windows broken. Unstable objects overturned. Pendulum clocks may stop.	0.011 to 0.029	4.0 to 4.9 (Light)
VI.	Felt by all, many frightened. Some heavy furniture moved; a few instances of fallen plaster. Damage slight.	0.03 to 0.2 (CNS Design Basis OBE = 0.1; CNS Design Basis SSE = 0.2)	5.0 to 5.9 (Moderate)
VII.	Damage negligible in buildings of good design and construction; slight to moderate in well-built ordinary structures; considerable damage in poorly built or badly designed structures; some chimneys broken.		
VIII.	Damage slight in specially designed structures; considerable damage in ordinary substantial buildings with partial collapse. Damage great in poorly built structures. Fall of chimneys, factory stacks, columns, monuments, walls. Heavy furniture overturned.	0.2 to 0.4	6.0 to 6.9 (Strong)
IX.	Damage considerable in specially designed structures; well-designed frame structures thrown out of plumb. Damage great in substantial buildings, with partial collapse. Buildings shifted off foundations.	0.4 to 0.5	7.0 to 7.9 (Major)
X.	Some well-built wooden structures destroyed; most masonry and frame structures destroyed with foundations. Rails bent.		
XI.	Few, if any (masonry), structures remain standing. Bridges destroyed. Rails bent greatly.	> 0.5	8.0 and higher (Great)
XII.	Damage total. Lines of sight and level are distorted. Objects thrown into the air.		

* The Richter magnitude and maximum acceleration (g) in the table above is provided for general comparison and description. Actual seismic intensity and effects will vary.

(continued on next page)

CNS Basis Reference(s):

1. Alarm Procedure 2.3_B-3, Panel B - Annunciator B-3/A-1.
2. Alarm Procedure 2.3_B-3, Panel B - Annunciator B-3/B-1.
3. Instrument Operating Procedure 4.12, Seismic Instrumentation.
4. Emergency Procedure 5.1Quake, Earthquake.
5. USAR Section II-5.2.4 and Table II-5-1.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting Vital Areas

EAL:

HA1.2 Alert

Tornado striking or high winds ≥ 100 mph resulting in **EITHER**:

Visible damage to **any** Table H-1 area structure containing safety systems or components

OR

Control Room indication of degraded performance of safety systems

Table H-1 Safe Shutdown Areas
• Reactor Building • Control Building • Service Water Pump Room • Diesel Generator Building • Cable Expansion Room

Mode Applicability:

All

(continued on next page)

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 indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occur on the basis of other EALs (e.g., System Malfunction).

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.

CNS Basis:

This threshold addresses events that may have resulted in Safe Shutdown Areas being subjected to forces (tornado or high winds > 100 mph) (Reference 6) beyond design limits. 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 (Reference 1-4, 7).

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.

The design wind pressure for the station and structures is 30 lb/ft^2 which is the equivalent of sustained winds up to 100 mph (Reference 6).

The Protected Area refers to the designated security area around the process buildings and is depicted in Technical Specifications Figure 4.1-1 (Reference 5).

The Critical Switchgear Rooms are a part of the Reactor Building.

(continued on next page)

CNS Basis Reference(s):

1. Site Services Procedure 1.1, Station Security.
2. CNS-FP-60, Fire Area Boundary Drawing Index.
3. Drawing CNS-EE-187, CNS Safe Shutdown Component Locations and Emergency Route Lighting - Site Plan.
4. Fire Safety Analysis Calculations.
5. Technical Specifications Figure 4.1-1, Site and Exclusion Area Boundaries and Low Population Zone.
6. USAR Section II-3.2.2.
7. USAR Section XII-2.1.2.1, Principal Class I Structures Required for Safe Shutdown.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting Vital Areas

EAL:

HA1.3 Alert

Main turbine failure-generated projectiles result in **EITHER**:
Visible damage to or penetration of **any** Table H-1 area structure containing safety systems or components

OR

Control Room indication of degraded performance of safety systems

Table H-1 Safe Shutdown Areas
• Reactor Building • Control Building • Service Water Pump Room • Diesel Generator Building • Cable Expansion Room

Mode Applicability:

All

(continued on next page)

NEI 99-01 Basis:

The 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 indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occur on the basis of other EALs (e.g., System Malfunction).

CNS Basis:

The main 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.

The list of Table H-1 areas includes all areas containing safety-related equipment, their controls, and their power supplies (Reference 1-5).

The Critical Switchgear Rooms are a part of the Reactor Building.

(continued on next page)

CNS Basis Reference(s):

1. Site Services Procedure 1.1, Station Security.
2. CNS-FP-60, Fire Area Boundary Drawing Index.
3. Drawing CNS-EE-187, CNS Safe Shutdown Component Locations and Emergency Route Lighting - Site Plan.
4. Fire Safety Analysis Calculations.
5. USAR Section XII-2.1.2.1, Principal Class I Structures Required for Safe Shutdown.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting Vital Areas

EAL:

HA1.4 Alert

Flooding in **any** Table H-1 area resulting in **EITHER**:

An electrical shock hazard that precludes access to operate or monitor safety equipment

OR

Control Room indication of degraded performance of safety systems

Table H-1 Safe Shutdown Areas
• Reactor Building • Control Building • Service Water Pump Room • Diesel Generator Building • Cable Expansion Room

Mode Applicability:

All

(continued on next page)

NEI 99-01 Basis:

The EAL escalates from HU1.4 in that the occurrence of the event has resulted in an electrical shock hazard precluding access to plant structures or areas containing equipment necessary for a safe shutdown or has caused damage to the safety systems in those structures evidenced by control indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occur on the basis of other EALs (e.g., System Malfunction).

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.

CNS Basis:

The internal flooding areas of concern are listed in Table H-1 (Reference 1-5).

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 this area was determined by an examination of piping systems in the area and also considered propagation of water from one area to another.

The accumulation of water resulting in a rising water level in the area constitutes flooding.

The Critical Switchgear Rooms are a part of the Reactor Building.

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CNS Basis Reference(s):

1. Site Services Procedure 1.1, Station Security.
2. CNS-FP-60, Fire Area Boundary Drawing Index.
3. Drawing CNS-EE-187, CNS Safe Shutdown Component Locations and Emergency Route Lighting - Site Plan.
4. Fire Safety Analysis Calculations.
5. USAR Section XII-2.1.2.1, Principal Class I Structures Required for Safe Shutdown.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting Vital Areas

EAL:

HA1.5 Alert

High river/forebay water level > 902' MSL

OR

Low river/forebay level < 865' MSL

Mode Applicability:

All

NEI 99-01 Basis:

Escalation to higher classifications occurs on the basis of other EALs (e.g., System Malfunction).

This EAL addresses other site specific 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.

CNS Basis:

HU1.5 covers high river/forebay water level conditions that could pose a significant threat to plant safety as well as low river/forebay water level conditions which may threaten operability of vital emergency plant cooling systems.

A river level of 902' requires reactor shutdown and represents the maximum possible (10,000 year) flood stage (Reference 3).

A river level of 865' MSL corresponds to the Safe Shutdown low river level and threatens availability of cooling systems and heat sink (Reference 1, 4, 5).

The forebay refers to the area between the east Intake Structure wall and the guide wall (Reference 2).

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CNS Basis Reference(s):

1. Emergency Procedure 5.1FLOOD, Flood.
2. USAR Section II-2.7.2.
3. USAR Section II-4.2.2.1.
4. USAR Section II-4.2.2.2.
5. USAR Section II-4.2.3.2.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 1 - Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting Vital Areas

EAL:

HA1.6 Alert

Vehicle crash resulting in **EITHER**:

Visible damage to **any** Table H-1 area structure containing safety systems or components

OR

Control Room indication of degraded performance of safety systems

Table H-1 Safe Shutdown Areas
• Reactor Building • Control Building • Service Water Pump Room • Diesel Generator Building • Cable Expansion Room

Mode Applicability:

All

(continued on next page)

NEI 99-01 Basis:

This EAL is based on the occurrence of a vehicle crash that has resulted in visible damage to plant structures or areas containing equipment necessary for a safe shutdown or has caused damage to the safety systems in those structures evidenced by control indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to higher classifications occur on the basis of other EALs (e.g., System Malfunction).

This EAL addresses vehicle crashes within the PROTECTED AREA that results 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.

CNS 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 (Reference 1-4, 6).

The Protected Area refers to the designated security area around the process buildings and is depicted in Technical Specifications Figure 4.1-1 (Reference 5).

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

The Critical Switchgear Rooms are a part of the Reactor Building.

(continued on next page)

CNS Basis Reference(s):

1. Site Services Procedure 1.1, Station Security.
2. CNS-FP-60, Fire Area Boundary Drawing Index.
3. Drawing CNS-EE-187, CNS Safe Shutdown Component Locations and Emergency Route Lighting - Site Plan.
4. Fire Safety Analysis Calculations.
5. Technical Specifications Figure 4.1-1, Site and Exclusion Area Boundaries and Low Population Zone.
6. USAR Section XII-2.1.2.1, Principal Class I Structures Required for Safe Shutdown.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 2 - Fire or Explosion

Initiating Condition: Fire within the Protected Area not extinguished within 15 minutes of detection or explosion within the Protected Area

EAL:

HU2.1 Unusual Event

Fire in **any** Table H-1 area **not** extinguished within 15 min. of Control Room notification or receipt of a valid Control Room alarm due to fire (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 Building
• Service Water Pump Room
• Diesel Generator Building
• Cable Expansion Room

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.

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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).

CNS 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.

The Critical Switchgear Rooms are a part of the Reactor Building.

CNS Basis Reference(s):

1. Site Services Procedure 1.1, Station Security.
2. CNS-FP-60, Fire Area Boundary Drawing Index.
3. Drawing CNS-EE-187, CNS Safe Shutdown Component Locations and Emergency Route Lighting - Site Plan.
4. Fire Safety Analysis Calculations.
5. USAR Section XII-2.1.2.1, Principal Class I Structures Required for Safe Shutdown.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 2 - Fire or Explosion

Initiating Condition: Fire within the Protected Area not extinguished within 15 minutes of detection or explosion within the Protected Area

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 capable of causing damage to permanent structures or equipment within the Protected Area.

No attempt is made to assess the actual magnitude of any 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 HA2.1.

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CNS Basis:

The Protected Area refers to the designated Security Area around the process buildings and is depicted in Technical Specifications Figure 4.1-1 (Reference 1).

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 (EAL Category F).

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

CNS Basis Reference(s):

1. Technical Specifications Figure 4.1-1, Site and Exclusion Area Boundaries and Low Population Zone.

Category: H - Hazards And Other Conditions Affecting Plant Safety

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 **EITHER**:

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 Building • Service Water Pump Room • Diesel Generator Building • Cable Expansion Room

Mode Applicability:

All

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.

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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.

CNS 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).

The Critical Switchgear Rooms are a part of the Reactor Building.

CNS Basis Reference(s):

1. Site Services Procedure 1.1, Station Security.
2. CNS-FP-60, Fire Area Boundary Drawing Index.
3. Drawing CNS-EE-187, CNS Safe Shutdown Component Locations and Emergency Route Lighting - Site Plan.
4. Fire Safety Analysis Calculations.
5. USAR Section XII-2.1.2.1, Principal Class I Structures Required for Safe Shutdown.

Category: H - Hazards And Other Conditions Affecting Plant Safety

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 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 EAL 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 HA3.1.

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CNS 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. 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. Administrative Procedure 0.36.6, Monitoring for Industrial Gases, may be used for help in assessing this EAL. Such review, however, does not constitute a departure from normal operations.

The release may have originated within the Site Boundary or it may have originated off-site and subsequently drifted onto the Site Boundary. Off-site 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.

At CNS there are various potential sources of atmospheric contamination. Some of these sources are:

- Inert gas used for oxygen exclusion (nitrogen).
- Combustion products.
- Carbon dioxide from fire extinguishing.
- Halon from fire extinguishing.
- Welding gases (enclosed areas).
- Vapors from painting (enclosed areas).
- Vapors from petroleum products.
- Hydrogen (OWC Hydrogen Gas Generation System, Generator Cooling System, batteries, and disassociation of water in the reactor).
- Asphyxiants and irritants, found most often in confined areas (water and oil storage tanks, open manholes).
- Methane from bacterial action (tanks and pits).

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Some of the gases which could affect normal plant operations under this EAL are:

- Carbon Monoxide - One of the most common asphyxiants encountered in industry. It is formed by the incomplete combustion of fuel containing carbon. It may be found in the vicinity of a fire or a leak in an exhaust system (flue gas or internal combustion engines).
- Oxygen - Oxygen has two fundamentally important properties: it supports combustion and it supports life. Since oxygen is necessary for life, it must be present in sufficient quantity. Oxygen deficiency occurs in confined spaces where the level of oxygen has been reduced below the limit to support life. Oxygen content in the air can become fatally low in a brief period of time. Some of the more common causes of this problem are oxidation of metals, bacterial action, combustion, and displacement by other gases. An enriched oxygen atmosphere will accelerate combustion.
- Hydrogen - Used in generator cooling. Hydrogen gas is also produced by the OWC Gas Generation System, located in the OWC Building, and subsequently injected into the Condensate System just upstream of the condensate booster pumps. Hydrogen is also produced by the disassociation of water from radiation in the reactor, which is seen in the off-gas. The presence of hydrogen will be especially significant in the Off-Gas and Augmented Off-Gas Systems. Hydrogen is also a by-product of battery charging. It is lighter than air so it will be found in pockets at the ceiling of enclosures.
- Argon - Commonly used during the welding of certain metals. It is denser than air so it will settle in pockets below the welding area.
- Carbon Dioxide - Used to fight fire. Being heavier than air, carbon dioxide will settle in pockets and displace oxygen.
- Halon - Used to fight fire in the Service Water Pump Room, Computer Room, and in the Simulator. Discharge of a Halon System will result in exceeding IDLH limits in the area. Discharge of the Service Water Pump Room or Computer Room Halon System should be classified under EAL HA3.1.
- Nitrogen - Used primarily to purge Primary Containment. Since it is approximately the same density as air, it can be dispersed by proper ventilation. Areas of poor ventilation may contain greater than expected concentrations of nitrogen and consequently may be deficient in oxygen.
- Combustible Gases and Vapors - Includes naturally occurring gases (such as methane and hydrogen gas) and the vapors of a large group of liquids which are used as fuels and solvents. Monitoring shall be required in fuel tanks and other areas where explosive mixtures may be present.

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- Hydrogen Sulfide - Classified as an irritant in low concentrations, but is even more toxic than carbon monoxide because it inflames the mucus membranes and results in the lungs filling with fluid. This colorless gas has a characteristic rotten egg odor, which renders the sense of smell ineffective. Hydrogen sulfide may be found in sewage treatment or wherever organic matter containing sulfur decomposes and shall be monitored constantly during work.
- Methane - The chief constituent of natural gas and is extremely explosive. It is non-toxic, but may reduce the oxygen content of an atmosphere, causing asphyxiation. Methane is often found in the vicinity of sanitary landfills and has been detected in tanks where bacterial action is taking place (i.e., reactor water cleanup and condensate phase separator tanks). It is lighter than air and tends to accumulate in high spots or pockets. This can present a dangerous situation in storage tanks or sewers where access is normally gained at the top of the confined area.
- Ethyl Benzene - Used primarily as an additive to diesel fuel. Acute exposure results in a local irritant effect on the skin and mucous membranes. Chronic exposure can lead to nervous system disorders and upper respiratory tract inflammation. Monitoring is required when entering a diesel fuel tank.
- Chlorine - Used in chemical treatment of Circulating Water and Service Water Systems. Chlorine gas can be recognized by its pungent, irritating odor, which is like the odor of bleach. Chlorine is not flammable but can react explosively with other chemicals such as turpentine or ammonia. Chlorine gas stays close to the ground and spreads rapidly. When chlorine gas comes in contact with moist human tissues, such as the eyes throat and lungs, an acid is produced that can damage these tissues.
- Chlorine Dioxide - This is a yellow to reddish-yellow manufactured gas which does not occur naturally in the environment. When added to water, chlorine dioxide forms chlorite ion, which is also a very reactive chemical. High levels of chlorine dioxide can be irritating to the nose, eyes, throat, and lungs.
- Hydrogen Chloride - This is a colorless to slightly yellowish gas with a pungent odor. On exposure to air, the gas forms dense white vapors due to condensation with atmospheric moisture. The vapor is corrosive and air concentrations above 5 ppm can cause irritation. When mixed with water or atmospheric moisture, a highly corrosive atmosphere is formed. The most common source of Hydrogen Chloride gas is from Muriatic (Hydrochloric) Acid.

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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.

CNS Basis Reference(s):

1. Administrative Procedure 0.36.6, Monitoring for Industrial Gases.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 3 - Toxic, Corrosive, Asphyxiant, And Flammable Gas

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

EAL:

HU3.2 Unusual Event
Recommendation by local, county, or state officials to evacuate or shelter site personnel based on an off-site 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 EAL 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 HA3.1.

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CNS Basis:

This EAL is based on the existence of an uncontrolled release originating off-site and local, county, or state officials have reported the need for evacuation or sheltering of site personnel. Off-site 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 off-site spills by using the Department of Transportation (DOT) Evacuation Tables for Selected Hazardous Materials in the DOT Emergency Response Guide for Hazardous Materials. If the evacuation area extends to any portion of the Owner Controlled Area, the EAL threshold is met.

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.

CNS Basis Reference(s):

1. Administrative Procedure 0.36.6, Monitoring for Industrial Gases.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 3 - Toxic, Corrosive, Asphyxiant, And Flammable Gas

Initiating Condition: Access to a Vital Area is prohibited due to release of toxic, corrosive, asphyxiant, or flammable gases which jeopardizes operation of operable equipment required to maintain safe operations or safely shutdown 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 shutdown the reactor (NOTE 7)

NOTE 7 – 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 Building • Service Water Pump Room • Diesel Generator Building • Cable Expansion Room

Mode Applicability:

All

NEI 99-01 Basis:

Gases in a Safe Shutdown Area can affect the ability to safely operate or safely shutdown the reactor.

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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 shutdown 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 gasses within a facility structure has the potential to affect safe operation of the plant by limiting either Operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Flammable gasses, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This EAL assumes concentrations of flammable gasses 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.

CNS 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 contain systems that are operated to establish or maintain safe shutdown (Reference 2-6).

The Critical Switchgear Rooms are a part of the Reactor Building.

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At CNS, there are various potential sources of atmospheric contamination. Some of these sources are:

- Inert gas used for oxygen exclusion (nitrogen).
- Combustion products.
- Carbon dioxide from fire extinguishing.
- Halon from fire extinguishing.
- Welding gases (enclosed areas).
- Vapors from painting (enclosed areas).
- Vapors from petroleum products.
- Hydrogen (OWC Hydrogen Gas Generation System, Generator Cooling System, batteries, and disassociation of water in the reactor).
- Asphyxiants and irritants found most often in confined areas (water and oil storage tanks, open manholes).
- Methane from bacterial action (tanks and pits).

Some of the gases which could affect normal plant operations under this EAL are:

- Carbon Monoxide - One of the most common asphyxiants encountered in industry. It is formed by the incomplete combustion of fuel containing carbon. It may be found in the vicinity of a fire or a leak in an exhaust system (flue gas or internal combustion engines).
- Oxygen - Oxygen has two fundamentally important properties: it supports combustion and it supports life. Since oxygen is necessary for life, it must be present in sufficient quantity. Oxygen deficiency occurs in confined spaces where the level of oxygen has been reduced below the limit to support life. Oxygen content in the air can become fatally low in a brief period of time. Some of the more common causes of this problem are oxidation of metals, bacterial action, combustion, and displacement by other gases. An enriched oxygen atmosphere will accelerate combustion.

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- Hydrogen - Used in generator cooling. Hydrogen gas is also produced by the OWC Gas Generation System, located in the OWC Building, and subsequently injected into the Condensate System just upstream of the condensate booster pumps. Hydrogen is also produced by the disassociation of water from radiation in the reactor, which is seen in the off-gas. The presence of hydrogen will be especially significant in the Off-Gas and Augmented Off-Gas Systems. Hydrogen is also a by-product of battery charging. It is lighter than air so it will be found in pockets at the ceiling of enclosures.
- Argon - Commonly used during the welding of certain metals. It is denser than air so it will settle in pockets below the welding area.
- Carbon Dioxide - Used to fight fire. Being heavier than air, carbon dioxide will settle in pockets and displace oxygen.
- Halon - Used to fight fire in the Service Water Pump Room, Computer Room, and in the Simulator. Discharge of a Halon System will result in exceeding IDLH limits in the area. Discharge of the Service Water Pump Room or Computer Room Halon System should be classified under this EAL.
- Nitrogen - Used primarily to purge Primary Containment. Since it is approximately the same density as air, it can be dispersed by proper ventilation. Areas of poor ventilation may contain greater than expected concentrations of nitrogen and consequently may be deficient in oxygen.
- Combustible Gases and Vapors - Includes naturally occurring gases (such as methane and hydrogen gas) and the vapors of a large group of liquids which are used as fuels and solvents. Monitoring shall be required in fuel tanks and other areas where explosive mixtures may be present.
- Hydrogen Sulfide - Classified as an irritant in low concentrations, but is even more toxic than carbon monoxide, because it inflames the mucus membranes and results in the lungs filling with fluid. This colorless gas has a characteristic rotten egg odor, which renders the sense of smell ineffective. Hydrogen sulfide may be found in sewage treatment or wherever organic matter containing sulfur decomposes and shall be monitored constantly during work.
- Methane - The chief constituent of natural gas and is extremely explosive. It is non-toxic, but may reduce the oxygen content of an atmosphere, causing asphyxiation. Methane is often found in the vicinity of sanitary landfills and has been detected in tanks where bacterial action is taking place (i.e., reactor water cleanup and condensate phase separator tanks). It is lighter than air and tends to accumulate in high spots or pockets. This can present a dangerous situation in storage tanks or sewers where access is normally gained at the top of the confined area.

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- Ethyl Benzene - Used primarily as an additive to diesel fuel. Acute exposure results in a local irritant effect on the skin and mucous membranes. Chronic exposure can lead to nervous system disorders and upper respiratory tract inflammation. Monitoring is required when entering a diesel fuel tank.
- Chlorine - Used in chemical treatment of Circulate Water and Service Water Systems. Chlorine gas can be recognized by its pungent, irritating odor, which is like the odor of bleach. Chlorine is not flammable but can react explosively with other chemicals such as turpentine or ammonia. Chlorine gas stays close to the ground and spreads rapidly. When chlorine gas comes in contact with moist human tissues, such as the eyes throat and lungs, an acid is produced that can damage these tissues.
- Chlorine Dioxide - This is a yellow to reddish-yellow manufactured gas which does not occur naturally in the environment. When added to water, chlorine dioxide forms chlorite ion, which is also a very reactive chemical. High levels of chlorine dioxide can be irritating to the nose, eyes, throat, and lungs.
- Hydrogen Chloride - This is a colorless to slightly yellowish gas with a pungent odor. On exposure to air, the gas forms dense white vapors due to condensation with atmospheric moisture. The vapor is corrosive and air concentrations above 5 ppm can cause irritation. When mixed with water or atmospheric moisture, a highly corrosive atmosphere is formed. The most common source of Hydrogen Chloride gas is from Muriatic (Hydrochloric) Acid.

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

CNS Basis Reference(s):

1. Administrative Procedure 0.36.6, Monitoring for Industrial Gases.
2. Site Services Procedure 1.1, Station Security.
3. CNS-FP-60, Fire Area Boundary Drawing Index.
4. Drawing CNS-EE-187, CNS Safe Shutdown Component Locations and Emergency Route Lighting - Site Plan.
5. Fire Safety Analysis Calculations.
6. USAR Section XII-2.1.2.1, Principal Class I Structures Required for Safe Shutdown.

Category: H - Hazards And Other Conditions Affecting Plant Safety

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 Security Shift Supervisor

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 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. Consideration should be given to upgrading the emergency response status and emergency classification level in accordance with the Physical Security Plan and Emergency Plan.

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1st Threshold

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

This threshold is based on the CNS Security and Safeguards Contingency Plan. Security Plans are based on guidance provided by NEI 03-12.

2nd Threshold

The second threshold is 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 Physical Security Plan.

3rd Threshold

The intent of this EAL threshold is to ensure that notifications for the aircraft threat are made in a timely manner and that off-site response organizations and plant personnel are at a state of heightened awareness regarding the credible threat.

This EAL is met when a plant receives information regarding an aircraft threat from 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 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.

Escalation to Alert emergency classification level via HA4.1 would be appropriate if the threat involves an airliner less than 30 minutes away from the plant.

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CNS Basis:

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

CNS Basis Reference(s):

1. CNS Security and Safeguards Contingency Plan.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 4 - Security

Initiating Condition: Hostile action within the Owner Controlled Area or airborne attack threat©[1]

EAL:

HA4.1 Alert

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

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.

These EAL thresholds address 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 on-site evacuation, dispersal, or sheltering).

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1st Threshold

This EAL 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 Owner Controlled Area. Those events are adequately addressed by other EALs.

Although nuclear plant Security Officers are well trained and prepared to protect against hostile action, it is appropriate for off-site response organizations to be notified and encouraged to begin activation (if they do not normally) to be better prepared should it be necessary to consider further actions.

2nd Threshold

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

The intent of this EAL threshold is to ensure that notifications for the airliner attack threat are made in a timely manner and that off-site response organizations 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 EAL threshold is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. 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 Alert.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner. The status and size of the plane may be provided by NORAD through the NRC.

If not previously notified by the NRC that the airborne hostile action was intentional, then it would be expected, although not certain, that notification by an appropriate Federal agency would follow. In this case, appropriate federal agency is intended to be NORAD, FBI, FAA, or NRC. However, the declaration should not be unduly delayed awaiting Federal notification.

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CNS Basis:

Reference is made to the Security Shift Supervisor because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred.

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

CNS Basis Reference(s):

1. CNS Security and Safeguards Contingency Plan.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 4 - Security

Initiating Condition: Hostile action within the Protected Area©[1]

EAL:

HS4.1 Site Area Emergency A hostile action is occurring or has occurred within the **Protected Area** as reported by the Security Shift Supervisor

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 off-site response organizations 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.

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Although nuclear plant Security Officers are well trained and prepared to protect against hostile action, it is appropriate for off-site response organizations to be notified and encouraged to begin preparations for public protective actions (if they do not normally) to be better prepared should it be necessary to consider further actions.

If not previously notified by NRC that the airborne hostile action was intentional, then it would be expected, although not certain, that notification by an appropriate Federal agency would follow. In this case, appropriate federal agency is intended to be NORAD, FBI, FAA, or NRC. However, the declaration should not be unduly delayed awaiting Federal notification.

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

CNS Basis:

Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs. Reference is made to the Security Shift Supervisor because this individual is the designated on-site person qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the CNS Security and Safeguards Contingency Plan (Reference 1).

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

CNS Basis Reference(s):

1. CNS Security and Safeguards Contingency Plan.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 4 - Security

Initiating Condition: Hostile action resulting in loss of physical control of the facility©[1]

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 a freshly off-loaded reactor core in pool

Mode Applicability:

All

NEI 99-01 Basis:

<u>1st Threshold</u>

This EAL threshold 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.

These safety functions are reactivity control (ability to shut down the reactor and keep it shut down), reactor water level (ability to cool the core), and decay heat removal (ability to maintain a 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. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions.

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

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2nd Threshold

This EAL threshold 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.

CNS Basis:

A freshly off-loaded reactor core in pool consists of recently discharged fuel that has been out of the reactor for less than 1 year (Reference 1).

CNS Basis Reference(s):

1. Nuclear Procedure 10.6, SFSP Fuel Storage Constraints (restricted access for B.5.b).
2. CNS Security and Safeguards Contingency Plan.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 5 - Control Room Evacuation

Initiating Condition: Control Room evacuation has been initiated

EAL:

HA5.1 Alert
Procedure 5.1ASD, Alternate Shutdown, or Procedure 5.4FIRE-S/D, Fire Induced 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.

CNS Basis:

Procedures 5.1ASD, Alternate Shutdown (Reference 1), and 5.4FIRE-S/D, Fire Induced Shutdown From Outside Control Room (Reference 2), provide the instructions for scramming the unit and maintaining RCS inventory from outside the Control Room. The Shift Manager (SM) determines if 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.

CNS Basis Reference(s):

1. Emergency Procedure 5.1ASD, Alternate Shutdown.
2. Emergency Procedure 5.4FIRE-S/D, Fire Induced Shutdown From Outside Control Room.

Category: H - Hazards And Other Conditions Affecting Plant Safety

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 15 minute time for transfer starts when the Control Room begins to be evacuated (not when Procedure 5.1ASD, Alternate Shutdown, is entered). The time interval is based on how quickly control must be re-established without core uncovery and/or core damage.

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).

The determination of whether or not control is established from outside the Control Room is based on Emergency Director (ED) judgment. The Emergency Director is expected to make a reasonable, informed judgment that control of the plant from the Alternate Shutdown Panels cannot be established within the 15 minute interval.

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Escalation of this emergency classification level, if appropriate, would be by Fission Product Barrier Degradation or Abnormal Rad Levels/Radiological Effluent EALs.

CNS Basis:

Procedures 5.1ASD, Alternate Shutdown (Reference 1), and 5.4FIRE-S/D, Fire Induced Shutdown From Outside Control Room (Reference 2), provide the instructions for scramming the unit and maintaining RCS inventory from outside the Control Room. The Shift Manager determines if 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.

The 15 minute criterion applies from the time that the Control Room begins to be evacuated.

CNS Basis Reference(s):

1. Emergency Procedure 5.1ASD, Alternate Shutdown.
2. Emergency Procedure 5.4FIRE-S/D, Fire Induced Shutdown From Outside Control Room.

Category: H - Hazards And Other Conditions Affecting Plant Safety

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 **EITHER**:

Events are in-progress or have occurred which indicate 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 off-site 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 Unusual Event emergency class.

(continued on next page)

CNS Basis:

The Emergency Director is the designated on-site individual having the responsibility and authority for implementing the CNS Emergency Plan. The Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (Reference 1).

CNS Basis Reference(s):

1. Emergency Plan for Cooper Nuclear Station, Section 5.0.

Category: H - Hazards And Other Conditions Affecting Plant Safety

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 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 Alert emergency class.

(continued on next page)

CNS Basis:

The Emergency Director is the designated on-site individual having the responsibility and authority for implementing the CNS Emergency Plan. The Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (Reference 2).

For the purposes of this EAL, the Site Boundary for CNS is a 1 mile radius around the plant.

CNS Basis Reference(s):

1. CNS Drawing DWG.2.2 (P3-A-45).
2. Emergency Plan for Cooper Nuclear Station, Section 5.0.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 6 - Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of 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 failures of plant functions needed for protection of the public **OR** Hostile action that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of; or 2) that prevent effective access to equipment needed for the protection of the public **Any** releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem 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 class description for Site Area Emergency.

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CNS Basis:

The Emergency Director is the designated on-site individual having the responsibility and authority for implementing the CNS Emergency Plan. The Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (Reference 2).

For the purposes of this EAL, the Site Boundary for CNS is a 1 mile radius around the plant.

CNS Basis Reference(s):

1. CNS Drawing DWG.2.2 (P3-A-45).
2. Emergency Plan for Cooper Nuclear Station, Section 5.0.

Category: H - Hazards And Other Conditions Affecting Plant Safety

Subcategory: 6 - Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of 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 Rem TEDE and 5 Rem 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 General Emergency class.

(continued on next page)

CNS Basis:

The Emergency Director is the designated on-site individual having the responsibility and authority for implementing the CNS Emergency Plan. The Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (Reference 2).

Releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the Site Boundary.

For the purposes of this EAL, the Site Boundary for CNS is a 1 mile radius around the plant.

CNS Basis Reference(s):

1. CNS Drawing DWG.2.2 (P3-A-45).
2. Emergency Plan for Cooper Nuclear Station, Section 5.0.

Category: S - System Malfunction

Subcategory: 1 - Loss of Power

Initiating Condition: Loss of **all** off-site AC power to critical buses for 15 minutes or longer

EAL:

SU1.1 Unusual Event
Loss of **all** off-site AC power (Table S-3) to critical 4160V Buses 1F and 1G 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 has exceeded or will likely exceed the applicable time.

Table S-3 AC Power Sources
Offsite
• Startup Station Service Transformer • Emergency Station Service Transformer • Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer (Note 8)
Onsite
• DG-1 • DG-2 • Main Generator

NOTE 8 – The time required to establish the backfeed is likely longer than the specified time interval. If off-normal plant conditions have already established the backfeed, its power to the safety-related buses may be considered an off-site power source.

(continued on next page)

Mode Applicability:

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

NEI 99-01 Basis:

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

CNS Basis:

The 4160V critical Buses 1F (Div. I) and 1G (Div. II) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following off-site sources of power: Figure S-1 illustrates the 4160V AC Distribution System (Reference 7, 8).

- Startup Transformer - The Startup Transformer provides a source of off-site AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of off-site AC power to the station whenever the main generator is off-line (< 160 MWe). The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the station Normal Transformer fails or that the main generator trips off-line.
- Emergency Transformer - The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by the 69 kV transmission line from OPPD. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.

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- Backfeeding power from the 345 kV line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. As mentioned in NOTE 8, the time required to establish the backfeed is likely longer than the 15 minute interval. If off-normal plant conditions have already established the backfeed, its power to the safety-related buses may be considered an off-site power source.

On-site power sources are the emergency diesel generators (DG-1 and DG-2) and the Main Generator.

If a power supply should have picked up a critical bus but failed to do so, that power supply should be considered unavailable until it has been successfully tied onto the bus. If the power supply can be tied to the bus within 15 minutes, then the power supply is to be considered available.

The Supplemental Diesel Generator (SDG) is not considered an on-site or an off-site emergency power supply and is not considered in classifications involving loss of power. The SBO Coping Time per Regulatory Guide 1.155 considers the impact of a SDG.

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

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CNS Basis Reference(s):

1. System Operating Procedure 2.2.15, Startup Transformer.
2. System Operating Procedure 2.2.16, Normal Station Service Transformer.
3. System Operating Procedure 2.2.17, Emergency Station Service Transformer.
4. System Operating Procedure 2.2.18, 4160V Auxiliary Power Distribution System.
5. System Operating Procedure 2.2.20, Standby AC Power System (Diesel Generator).
6. Emergency Procedure 5.3SBO, Station Blackout.
7. BR 3001, One Line Diagram.
8. BR 3002, Sheet 1.
9. NC43456, One Line Switching Diagram 161kV Substation.
10. Enercon Services, Inc. Report No. NPP1-PR-01, Station Blackout Coping Assessment for Cooper Nuclear Station, Revision 2.

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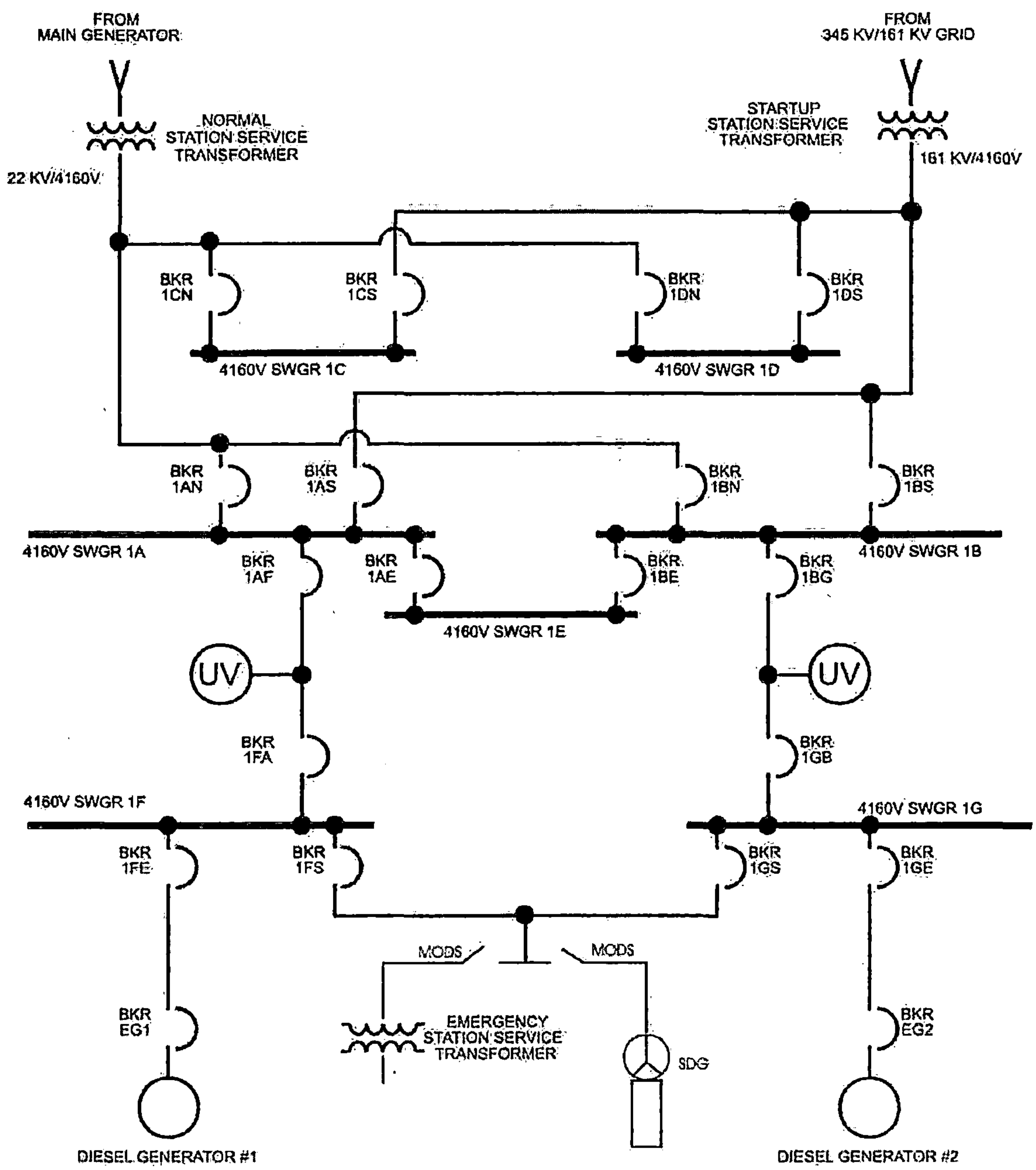
Figure S-1: 4160V AC Distribution System



Category: S - System Malfunction

Subcategory: 1 - Loss of Power

Initiating Condition: AC power capability to critical 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 critical buses

EAL:

SA1.1 Alert

AC power capability to critical 4160V Buses 1F and 1G reduced to a single power source (Table S-3) for ≥ 15 min. (NOTE 3) such that **any** additional single failure would result in loss of **all** AC power to emergency buses

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 Station Service Transformer • Emergency Station Service Transformer • Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer (Note 8)
Onsite
• DG-1 • DG-2 • Main Generator

(continued on next page)

NOTE 8 – The time required to establish the backfeed is likely longer than the specified time interval. If off-normal plant conditions have already established the backfeed, its power to the safety-related buses may be considered an off-site power source.

Mode Applicability:

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

NEI 99-01 Basis:

This EAL is intended to provide an escalation from IC SU1.1, "Loss of all off-site AC power to critical buses for 15 minutes or longer".

The condition indicated by this EAL is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a loss of all AC power to the critical buses. This condition could occur due to a loss of off-site power with a concurrent failure of all but one emergency generator to supply power to its critical bus. Another related condition could be the loss of all off-site power and loss of on-site emergency generators with only one train of critical buses being backfed from the unit main generator or the loss of on-site emergency generators with only one train of critical buses being backfed from off-site power. The subsequent loss of this single power source would escalate the event to a Site Area Emergency in accordance with SS1.1.

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

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CNS Basis:

The 4160V critical Buses 1F (Div. I) and 1G (Div. II) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following off-site sources of power: Figure S-1 illustrates the 4160V AC Distribution System (Reference 7, 8).

- Startup Transformer - The Startup Transformer provides a source of off-site AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of off-site AC power to the station whenever the main generator is off-line (< 160 MWe). The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the station Normal Transformer fails or that the main generator trips off-line.
- Emergency Transformer - The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by the 69 kV transmission line from OPPD. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.
- Backfeeding power from the 345 kV line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. As mentioned in NOTE 8, the time required to establish the backfeed is likely longer than the 15 minute interval. If off-normal plant conditions have already established the backfeed, its power to the safety-related buses may be considered an off-site power source.

On-site power sources are the emergency diesel generators (DG-1 and DG-2) and the Main Generator.

If a power supply should have picked up a critical bus but failed to do so, that power supply should be considered unavailable until it has been successfully tied onto the bus. If the power supply can be tied to the bus within 15 minutes, then the power supply is to be considered available.

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The Supplemental Diesel Generator (SDG) is not considered an on-site or an off-site emergency power supply and is not considered in classifications involving loss of power. The SBO Coping Time per Regulatory Guide 1.155 considers the impact of a SDG.

The 15 minute interval was selected as a threshold to exclude transient or momentary power losses. If the capability of a second source of emergency bus power is not restored within 15 minutes, an Alert is declared under this EAL.

CNS Basis Reference(s):

1. System Operating Procedure 2.2.15, Startup Transformer.
2. System Operating Procedure 2.2.16, Normal Station Service Transformer.
3. System Operating Procedure 2.2.17, Emergency Station Service Transformer.
4. System Operating Procedure 2.2.18, 4160V Auxiliary Power Distribution System.
5. System Operating Procedure 2.2.20, Standby AC Power System (Diesel Generator).
6. Emergency Procedure 5.3SBO, Station Blackout.
7. BR 3001, One Line Diagram.
8. BR 3002, Sheet 1.
9. NC43456, One Line Switching Diagram 161kV Substation.
10. Enercon Services, Inc. Report No. NPP1-PR-01, Station Blackout Coping Assessment for Cooper Nuclear Station, Revision 2.

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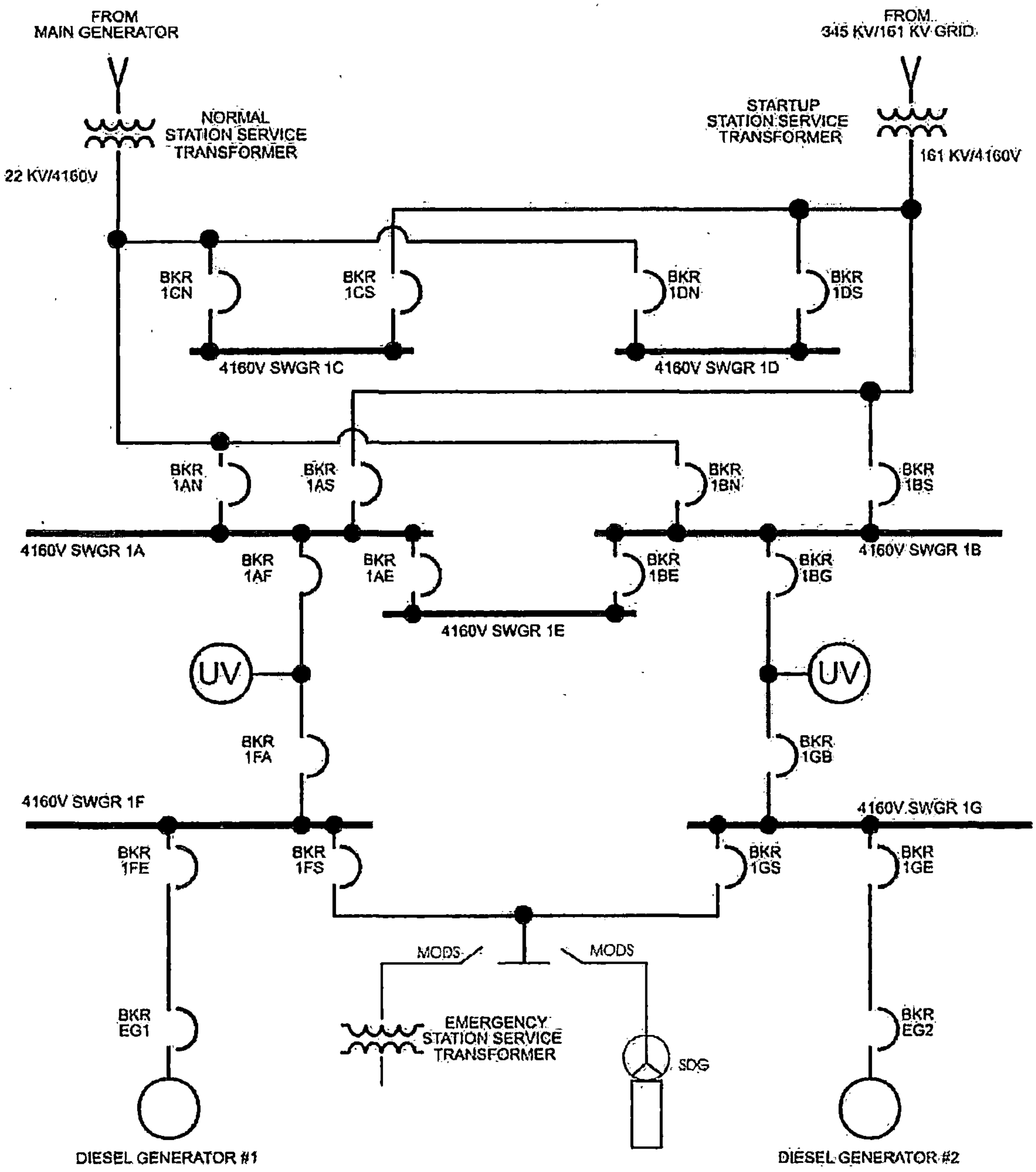
Figure S-1: 4160V AC Distribution System



Category: S - System Malfunction

Subcategory: 1 - Loss of Power

Initiating Condition: Loss of **all** off-site and **all** on-site AC power to critical buses for 15 minutes or longer

EAL:

SS1.1 Site Area Emergency

Loss of **all** off-site and **all** on-site AC power (Table S-3) to critical 4160V Buses 1F and 1G 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 Station Service Transformer • Emergency Station Service Transformer • Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer (Note 8)
Onsite
• DG-1 • DG-2 • Main Generator

NOTE 8 – The time required to establish the backfeed is likely longer than the specified time interval. If off-normal plant conditions have already established the backfeed, its power to the safety-related buses may be considered an off-site power source.

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Mode Applicability:

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

NEI 99-01 Basis:

Loss of all AC power to critical 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 critical 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 off-site power.

Escalation to General Emergency is via Fission Product Barrier Degradation or IC SG1, "Prolonged loss of all off-site power and prolonged loss of all on-site AC power".

CNS Basis:

The 4160V critical Buses 1F (Div. I) and 1G (Div. II) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following off-site sources of power: Figure S-1 illustrates the 4160V AC Distribution System (Reference 7, 8).

- Startup Transformer - The Startup Transformer provides a source of off-site AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of off-site AC power to the station whenever the main generator is off-line (< 160 MWe). The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the station Normal Transformer fails or that the main generator trips off-line.

(continued on next page)

- Emergency Transformer - The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by the 69 kV transmission line from OPPD. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.
- Backfeeding power from the 345 kV line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. As mentioned in NOTE 8, the time required to establish the backfeed is likely longer than the 15 minute interval. If off-normal plant conditions have already established the backfeed, its power to the safety-related buses may be considered an off-site power source.

On-site power sources are the emergency diesel generators (DG-1 and DG-2) and the Main Generator.

If a power supply should have picked up a critical bus but failed to do so, that power supply should be considered unavailable until it has been successfully tied onto the bus. If the power supply can be tied to the bus within 15 minutes, then the power supply is to be considered available.

The Supplemental Diesel Generator (SDG) is not considered an on-site or an off-site emergency power supply and is not considered in classifications involving loss of power. The SBO Coping Time per Regulatory Guide 1.155 considers the impact of a SDG.

This EAL is the hot condition equivalent of the cold condition loss of all AC power EAL CA1.1. When in Cold Shutdown, Refueling, or Defueled Mode, the event can be classified as an Alert because of the significantly reduced decay heat, lower temperature and pressure, increasing the time to restore one of the critical buses, relative to that existing when in hot conditions.

(continued on next page)

CNS Basis Reference(s):

1. System Operating Procedure 2.2.15, Startup Transformer.
2. System Operating Procedure 2.2.16, Normal Station Service Transformer.
3. System Operating Procedure 2.2.17, Emergency Station Service Transformer.
4. System Operating Procedure 2.2.18, 4160V Auxiliary Power Distribution System.
5. System Operating Procedure 2.2.20, Standby AC Power System (Diesel Generator).
6. Emergency Procedure 5.3SBO, Station Blackout.
7. BR 3001, One Line Diagram.
8. BR 3002, Sheet 1.
9. NC43456, One Line Switching Diagram 161kV Substation.
10. Enercon Services, Inc. Report No. NPP1-PR-01, Station Blackout Coping Assessment for Cooper Nuclear Station, Revision 2.

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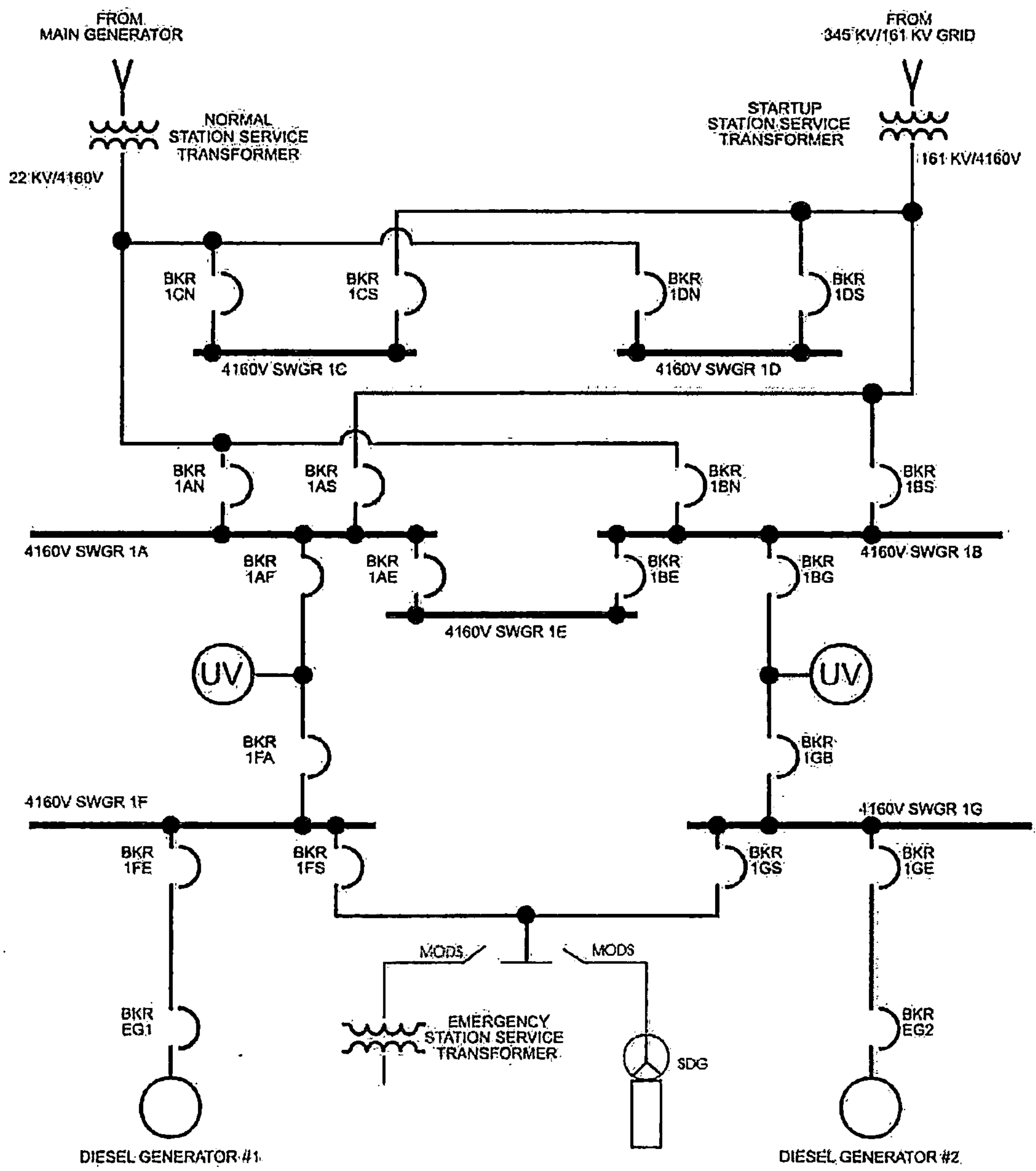
Figure S-1: 4160V AC Distribution System



Category: S - System Malfunction

Subcategory: 1 - Loss of Power

Initiating Condition: Prolonged loss of **all** off-site and **all** on-site AC power to critical emergency buses

EAL:

SG1.1 General Emergency
Loss of **all** off-site and **all** on-site AC power (Table S-3) to critical 4160V Buses 1F and 1G, **AND EITHER:**
Restoration of at least one emergency bus in < 4 hours is **not** likely
OR
RPV level **cannot** be restored and maintained > -158 in. or **cannot** be determined

Table S-3 AC Power Sources
Offsite
• Startup Station Service Transformer • Emergency Station Service Transformer • Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer (Note 8)
Onsite
• DG-1 • DG-2 • Main Generator

NOTE 8 – The time required to establish the backfeed is likely longer than the specified time interval. If off-normal plant conditions have already established the backfeed, its power to the safety-related buses may be considered an off-site power source.

(continued on next page)

Mode Applicability:

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

NEI 99-01 Basis:

Loss of all AC power to critical 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 critical 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 loss of all critical bus AC power, 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 critical bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

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

Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly (s)he may need to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of fission product barriers is imminent?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers.

(continued on next page)

CNS Basis:

The 4160V Critical Buses 1F (Div. I) and 1G (Div. II) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following off-site sources of power: Figure S-1 illustrates the 4160V AC Distribution System (Reference 7, 8).

- Startup Transformer - The Startup Transformer provides a source of off-site AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of off-site AC power to the station whenever the main generator is off-line (< 160 MWe). The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the station Normal Transformer fails or that the main generator trips off-line.
- Emergency Transformer - The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by the 69 kV transmission line from OPPD. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.
- Backfeeding power from the 345 kV line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. As mentioned in NOTE 8, the time required to establish the backfeed is likely longer than the 4 hour interval. If off-normal plant conditions have already established the backfeed; however, its power to the safety-related buses may be considered an off-site power source.

On-site power sources are the emergency diesel generators (DG-1 and DG-2) and the Main Generator.

If a power supply should have picked up a critical bus but failed to do so, that power supply should be considered unavailable until it has been successfully tied onto the bus.

4 hours is the CNS Station Blackout Coping Analysis time (Reference 11, 13).

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The Supplemental Diesel Generator (SDG) is not considered an on-site or an off-site emergency power supply and is not considered in classifications involving loss of power. The SBO Coping Time per Regulatory Guide 1.155 considers the impact of a SDG.

Indication of continuing core cooling degradation is manifested by a RPV level instrument reading of < -158 in. (RPV level is below the top of active fuel). When RPV level is at or above the top of active fuel, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below the top of active fuel, 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 the top of active fuel, 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-2B, RPV Flooding, or EOP-7B, RPV Flooding (Failure-to-Scram). RPV water level indication provides the primary means of knowing if 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-2B/7B 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.

(continued on next page)

CNS Basis Reference(s):

1. System Operating Procedure 2.2.15, Startup Transformer.
2. System Operating Procedure 2.2.16, Normal Station Service Transformer.
3. System Operating Procedure 2.2.17, Emergency Station Service Transformer.
4. System Operating Procedure 2.2.18, 4160V Auxiliary Power Distribution System.
5. System Operating Procedure 2.2.20, Standby AC Power System (Diesel Generator).
6. Emergency Procedure 5.3SBO, Station Blackout.
7. BR 3001, One Line Diagram.
8. BR 3002, Sheet 1.
9. NC43456, One Line Switching Diagram 161kV Substation.
10. EOP-2B RPV Flooding.
11. EOP-7B RPV Flooding (Failure-to-Scram).
12. Enercon Services, Inc. Report No. NPP1-PR-01, Station Blackout Coping Assessment for Cooper Nuclear Station, Revision 2.
13. NEDC 97-089.
14. USAR Section VIII-6.2.7.

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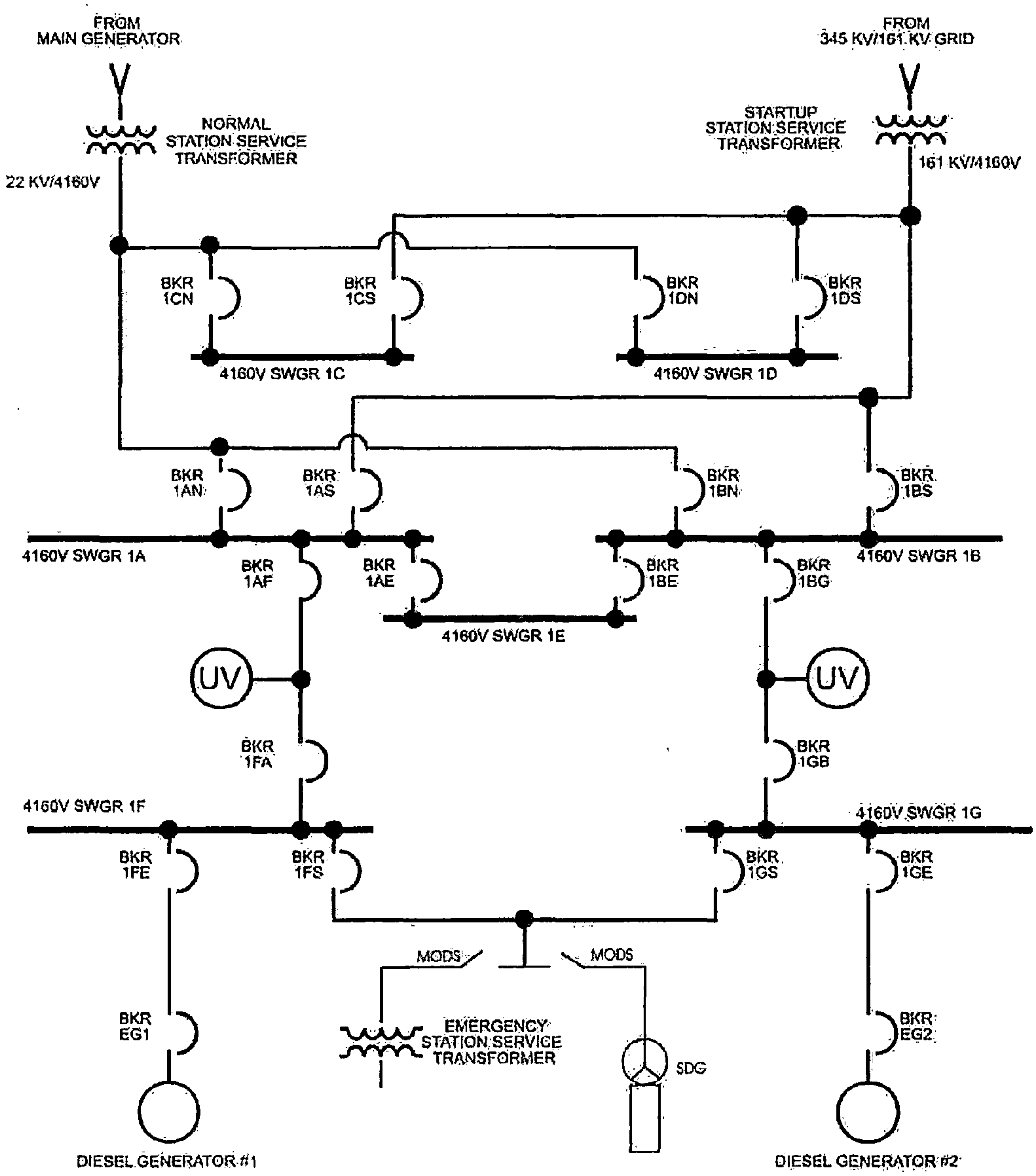
Figure S-1: 4160V AC Distribution System



Category: S - System Malfunction

Subcategory: 2 - ATWS/Criticality

Initiating Condition: Inadvertent criticality

EAL:

SU2.1	Unusual Event
An unplanned sustained positive period observed on nuclear instrumentation	

Mode Applicability:

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 an Unusual Event 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.

CNS Basis:

SRM A-D period Meters NMS-I-44A-D on Panel 9-5 identify this condition as well as Panel 9-5 amber light and SRM Period (< 50 sec.) Annunciator 9-5-1/F-8 (Reference 1, 2). However, a SRM period alarm caused by SRM channel noise does not result in entry into this EAL (Reference 1).

CNS Basis Reference(s):

1. Alarm Procedure 2.3_9-5-1, Panel 9-5 - Annunciator 9-5-1, F-8, SRM Period.
2. Instrument Operating Procedure 4.1.1, Source Range Monitoring System.

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

AND

Manual actions taken at the reactor control console (NOTE 5) successfully shut down the reactor as indicated by reactor power < 3%

NOTE 5 – Manual scram methods for EAL SA2.1 and EAL SS2.1 are the following:

- Reactor Scram pushbuttons.
- Reactor Mode switch in SHUTDOWN.
- Manual or auto actuation of ARI.

Mode Applicability:

1 - Power Operation, 2 - Startup

NEI 99-01 Basis:

Manual scram (trip) 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.

(continued on next page)

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.

CNS Basis:

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function. A reactor scram is automatically initiated by the Reactor Protection System (RPS) when certain continuously monitored parameters exceed predetermined setpoints. A reactor scram may be the result of manual or automatic action in response to any of the following parameters (Reference 3):

- APRM Fixed Neutron Flux - High.
- APRM Fixed Neutron Flux - High (Setdown).
- APRM Flow Biased - High.
- IRM - High.
- Reactor Steam Dome Pressure - High.
- Reactor Vessel Water Level Low - Level-3.
- Turbine Stop Valve Closure.
- Turbine Control Valve Fast Closure.
- MSIV Closure.
- Scram Discharge Volume (SDV) Level - High.
- Drywell Pressure - High.

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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% (Reference 1, 2).

The significance of the second condition of this EAL 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, Procedure 2.1.5 prescribe insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved (Reference 1). 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.

This EAL is not applicable if a manual scram is initiated and no RPS setpoints are exceeded. Taking the mode switch to shutdown 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. Methods of inserting a manual scram are limited to those that can be taken rapidly at the reactor control console (Panel 9-5) and include:

- Both manual Reactor Scram pushbuttons.
- Reactor Mode switch in SHUTDOWN.
- Manual or auto actuation of ARI.

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Auto actuation of ARI is included in the list of methods because the Operator, by procedure, always ensures actuation of ARI has occurred if the ARI actuation setpoints are exceeded. This means action to depress the ARI pushbuttons is taken if the automatic ARI actuation setpoints are exceeded but failed to actuate. If ARI properly actuates automatically, the ARI pushbuttons are not depressed. Reactor shutdown achieved by use of the alternate rod insertion methods listed in Procedure 5.8.3 do not constitute a successful manual scram (Reference 2).

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% (Reference 1, 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 if 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, then 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.

CNS Basis Reference(s):

1. General Operating Procedure 2.1.5, Reactor Scram.
2. Emergency Procedure 5.8.3, Alternate Rod Insertion Methods.
3. AMP-TBD00, Appendix B, Step RC/Q-4.
4. USAR Table VII-2-2.

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

AND

Manual actions taken at the reactor control console (NOTE 5) do **not** shut down the reactor as indicated by reactor power ≥ 3%

NOTE 5 – Manual scram methods for EAL SA2.1 and EAL SS2.1 are the following:

- Reactor Scram pushbuttons.
- Reactor Mode switch in SHUTDOWN.
- Manual or auto actuation of ARI.

Mode Applicability:

1 - Power Operation, 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. 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) at which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

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Manual scram actions are not considered successful if action away from the reactor control console is required to scram the reactor. This EAL is still applicable even if actions taken away from the reactor control console are successful in shutting the reactor down 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.

CNS 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. Methods of inserting a manual scram are limited to those that can be taken rapidly at the reactor control console (Panel 9-5) and include (Reference 1):

- Both manual Reactor Scram pushbuttons.
- Reactor Mode switch in SHUTDOWN.
- Manual or auto actuation of ARI.

Auto actuation of ARI is included in the list of methods because the Operator, by procedure, always ensures actuation of ARI has occurred if the ARI actuation setpoints are exceeded. This means action to depress the ARI pushbuttons is taken if the automatic ARI actuation setpoints are exceeded but failed to actuate. If ARI properly actuates automatically, the ARI pushbuttons are not depressed. Reactor shutdown achieved by use of the alternate rod insertion methods listed in Emergency Procedure 5.8.3 do not constitute a successful manual scram (Reference 2).

The APRM downscale trip setpoint (3%) 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 if reactor power is greater than 3% power (Reference 1, 3).

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

CNS Basis Reference(s):

1. General Operating Procedure 2.1.5, Reactor Scram.
2. Emergency Procedure 5.8.3, Alternate Rod Insertion Methods.
3. AMP-TBD00, Appendix B, Step RC/Q-4.

Category: S - System Malfunction

Subcategory: 2 - ATWS/Criticality

Initiating Condition: Automatic scram and **all** manual actions fail to shut down the reactor and indication of an extreme challenge to the ability to cool the core exists

EAL:

SG2.1 General Emergency

Automatic and **all** manual scrams were **not** successful

AND

Reactor power ≥ 3%

AND EITHER of the following exist or have occurred due to continued power generation:

RPV level **cannot** be restored and maintained > -183 in. or **cannot** be determined

OR

Average torus water temperature and RPV pressure **cannot** be maintained within the Heat Capacity Temperature Limit (EOP/SAG Graph 7)

Mode Applicability:

1 - Power Operation, 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.

The extreme challenge to the ability to cool the core is intended to mean that the reactor vessel water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level as described in the EOP bases.

Considerations include inability to remove heat via the main condenser or via the suppression pool or torus (e.g., due to high pool water temperature).

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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 off-site intervention time.

CNS 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.

Reactor shutdown achieved by use of the alternate rod insertion methods listed in Procedure 5.8.3 are credited as a successful manual scram provided 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 (Reference 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 if reactor power is > 3% power (Reference 1, 3).

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.

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Indication that core cooling is extremely challenged is manifested by inability to restore and maintain RPV water level above -183 in. (or cannot be determined). -183 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 (Reference 5). 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.

A threshold prescribing declaration when a parameter value cannot be maintained above a specified limit does not require immediate action simply because the current value is below the limit (-183 inches), **but does not permit extended operation below the limit**. Because the systems used to restore water level have different volume and discharge pressure capabilities, and pressure control methods also have varying rates that RPV pressure can be modified to support injection, the threshold must be considered reached as soon as it is apparent that the limit cannot be attained within a reasonable amount of time. Determination of inability to restore and maintain RPV level is based on actions driven by EOPs to restore level.

When RPV level cannot be determined, EOPs require entry to EOP-7B, RPV Flooding (Failure-to-Scram). RPV water level indication provides the primary means of knowing if 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-7B 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.

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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-3A, Primary Containment Control, Step SP/T-5, is reached (Reference 4). This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature.

CNS Basis Reference(s):

1. General Operating Procedure 2.1.5, Reactor Scram.
2. Emergency Procedure 5.8.3, Alternate Rod Insertion Methods.
3. AMP-TBD00 Appendix B, Step RC/Q-4.
4. EOP-3A, Primary Containment Control.
5. NEDC 97-090J.

Category: S - System Malfunction

Subcategory: 3 - Inability to Reach Shutdown Conditions

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

EAL:

SU3.1 Unusual Event Plant is **not** brought to required operating mode within Technical Specifications LCO action statement time

Mode Applicability:

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

NEI 99-01 Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required Shutdown Mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the site Technical Specifications requires a 4 hour report under 10CFR50.72(b) non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate Unusual Event is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of an Unusual Event is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed.

Other required Technical Specification shutdowns that involve precursors to more serious events are addressed by other EALs.

CNS Basis:

None

CNS Basis Reference(s):

1. Technical Specifications.

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 Control Room Panels 9-3, 9-4, 9-5, and C 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 - Power Operation, 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.

"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.

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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 Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10CFR50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on SU3.1, Inability to Reach Required Shutdown Within Technical Specification Limits.

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.

CNS Basis:

The availability of computer-based monitoring capability (i.e., PMIS, SPDS) is not a factor at the Unusual Event emergency classification level. Safety system annunciation and indication considered in this EAL is found on Control Room Panels 9-3, 9-4, 9-5, and C. The other annunciators and indicators are important to plant operation but are not important to safety (Reference 1-14).

(continued on next page)

CNS Basis Reference(s):

1. System Operating Procedure 2.2.64, Annunciator System.
2. Alarm Procedure 2.3_9-3-1, Panel 9-3 - Annunciator 9-3-1.
3. Alarm Procedure 2.3_9-3-2, Panel 9-3 - Annunciator 9-3-2.
4. Alarm Procedure 2.3_9-3-3, Panel 9-3 - Annunciator 9-3-3.
5. Alarm Procedure 2.3_9-4-1, Panel 9-4 - Annunciator 9-4-1.
6. Alarm Procedure 2.3_9-4-2, Panel 9-4 - Annunciator 9-4-2.
7. Alarm Procedure 2.3_9-4-3, Panel 9-4 - Annunciator 9-4-3.
8. Alarm Procedure 2.3_9-5-1, Panel 9-5 - Annunciator 9-5-1.
9. Alarm Procedure 2.3_9-5-2, Panel 9-5 - Annunciator 9-5-2.
10. Alarm Procedure 2.3_C-1, Panel C - Annunciator C-1.
11. Alarm Procedure 2.3_C-2, Panel C - Annunciator C-2.
12. Alarm Procedure 2.3_C-3, Panel C - Annunciator C-3.
13. Alarm Procedure 2.3_C-4, Panel C - Annunciator C-4.
14. Abnormal Procedure 2.4ANN, Annunciator Failure.

Category: S - System Malfunction

Subcategory: 4 - Instrumentation

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 Control Room Panels 9-3, 9-4, 9-5, and C (NOTE 3) for ≥ 15 min. (NOTE 3)

AND EITHER:

Any significant transient is in-progress, Table S-1

OR

Compensatory indications are unavailable

<u>NOTE</u> 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%

Mode Applicability:

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

(continued on next page)

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 Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant 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 Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10CFR50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on EAL SU3.1, Inability to Reach Required Shutdown Within Technical Specification Limits.

"Compensatory indications" in this context includes computer based information such as PMIS/SPDS. 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.

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.

(continued on next page)

CNS Basis:

PMIS and SPDS serve as redundant compensatory indicators which may be utilized in lieu of normal Control Room indicators. Safety system annunciation and indication considered in this EAL is found on Control Room Panels 9-3, 9-4, 9-5, and C. The other annunciators and indicators are important to plant operation but are not important to safety (Reference 1-14).

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

CNS Basis Reference(s):

1. System Operating Procedure 2.2.64, Annunciator System.
2. Alarm Procedure 2.3_9-3-1, Panel 9-3 - Annunciator 9-3-1.
3. Alarm Procedure 2.3_9-3-2, Panel 9-3 - Annunciator 9-3-2.
4. Alarm Procedure 2.3_9-3-3, Panel 9-3 - Annunciator 9-3-3.
5. Alarm Procedure 2.3_9-4-1, Panel 9-4 - Annunciator 9-4-1.
6. Alarm Procedure 2.3_9-4-2, Panel 9-4 - Annunciator 9-4-2.
7. Alarm Procedure 2.3_9-4-3, Panel 9-4 - Annunciator 9-4-3.
8. Alarm Procedure 2.3_9-5-1, Panel 9-5 - Annunciator 9-5-1.
9. Alarm Procedure 2.3_9-5-2, Panel 9-5 - Annunciator 9-5-2.
10. Alarm Procedure 2.3_C-1, Panel C - Annunciator C-1.
11. Alarm Procedure 2.3_C-2, Panel C - Annunciator C-2.
12. Alarm Procedure 2.3_C-3, Panel C - Annunciator C-3.
13. Alarm Procedure 2.3_C-4, Panel C - Annunciator C-4.
14. Abnormal Procedure 2.4ANN, Annunciator Failure.

Category: S - System Malfunction

Subcategory: 4 - Instrumentation

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 Control Room Panels 9-3, 9-4, 9-5, and C for ≥ 15 min. (NOTE 3)

AND

Any significant transient is in-progress, Table S-1

AND

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%

Mode Applicability:

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

(continued on next page)

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 Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant 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 Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10CFR50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on SU3.1, Inability to Reach Required Shutdown Within Technical Specification 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 includes computer based information such as PMIS/SPDS. This should include all computer systems available for this use depending on specific plant design and subsequent retrofits.

(continued on next page)

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

CNS Basis:

The availability of computer-based monitoring capability (i.e., PMIS, SPDS) is a factor at the Site Area Emergency classification level because they are compensatory non-alarming indication. Safety system annunciation and indication considered in this EAL is found on Control Room Panels 9-3, 9-4, 9-5, and C. The other annunciators and indicators are important to plant operation but are not important to safety (Reference 1-14).

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

Due to the limited number of safety systems in operation during Cold Shutdown, Refueling and Defueled Modes, this EAL is not applicable during these modes of operation.

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CNS Basis Reference(s):

1. System Operating Procedure 2.2.64, Annunciator System.
2. Alarm Procedure 2.3_9-3-1, Panel 9-3 - Annunciator 9-3-1.
3. Alarm Procedure 2.3_9-3-2, Panel 9-3 - Annunciator 9-3-2.
4. Alarm Procedure 2.3_9-3-3, Panel 9-3 - Annunciator 9-3-3.
5. Alarm Procedure 2.3_9-4-1, Panel 9-4 - Annunciator 9-4-1.
6. Alarm Procedure 2.3_9-4-2, Panel 9-4 - Annunciator 9-4-2.
7. Alarm Procedure 2.3_9-4-3, Panel 9-4 - Annunciator 9-4-3.
8. Alarm Procedure 2.3_9-5-1, Panel 9-5 - Annunciator 9-5-1.
9. Alarm Procedure 2.3_9-5-2, Panel 9-5 - Annunciator 9-5-2.
10. Alarm Procedure 2.3_C-1, Panel C - Annunciator C-1.
11. Alarm Procedure 2.3_C-2, Panel C - Annunciator C-2.
12. Alarm Procedure 2.3_C-3, Panel C - Annunciator C-3.
13. Alarm Procedure 2.3_C-4, Panel C - Annunciator C-4.
14. Abnormal Procedure 2.4ANN, Annunciator Failure.

Category: S - System Malfunction

Subcategory: 5 - Fuel Clad Degradation

Initiating Condition: Fuel clad degradation

EAL:

SU5.1 Unusual Event
SJAE monitor > 1.58E+3 mR/hr

Mode Applicability:

1 - Power Operation, 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 site-specific radiation monitor readings that provide indication of a degradation of fuel clad integrity.

CNS Basis:

Steam Jet Air Ejectors (SJAEs) remove all non-condensable gases from the condensers including air in-leakage and disassociated products originating in the reactor and exhausts them to the off-gas holdup volume. A rise in off-gas activity could therefore indicate damage to the fuel cladding, a potential degradation in the level of safety of the plant, and a potential precursor of more serious problems. The Technical Specification allowable limit is ≤ 1 Ci/sec. The SJAE monitor Hi-Hi radiation setpoint is set at 50% of the instantaneous release limit and represents approximately 0.1% fuel cladding damage. The SJAE monitor Hi-Hi radiation setpoint has been selected because it is operationally significant and is readily recognizable by the Control Room Operating Staff (Reference 2-6). The Off-Gas System isolates after a 15 minute time delay (Reference 2, 3).

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In the Hot Modes, a steam source is available from which non-condensables can be separated for processing by the Off-Gas System. The Cold Shutdown, Refueling, and Defueled Modes do not afford a transfer mechanism from which the off-gas radiation monitors can draw a valid sample. The radiation monitors lose a valid sample source when the air ejectors are not in service (Reference 2, 4, 5).

CNS Basis Reference(s):

1. System Operating Procedure 2.2.55, Main Condenser Gas Removal System.
2. Alarm Procedure 2.3_9-4-1, Panel 9-4 - Annunciator 9-4-1, C-4, OFFGAS TIMER INITIATED.
3. Alarm Procedure 2.3_9-4-1, Panel 9-4 - Annunciator 9-4-1, C-5, OFFGAS HIGH RAD.
4. Abnormal Procedure 2.4OG, Off-Gas Abnormal.
5. Emergency Procedure 5.2FUEL, Fuel Failure.
6. NEDC 02-004, Estimation of the Steam Jet Air Ejector Radiation Monitor, RMP-RM-150A(B), Readings Following a 1% Fuel Clad Release (Degraded Core) in the Reactor Coolant System.
7. Technical Specification LCO 3.7.5, Air Ejector Off-Gas.

Category: S - System Malfunction

Subcategory: 5 - Fuel Clad Degradation

Initiating Condition: Fuel clad degradation

EAL:

SU5.2	Unusual Event
Coolant activity ≥ 4.0 μCi/gm dose equivalent I-131	

Mode Applicability:

1 - Power Operation, 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.

CNS 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 Specification LCO 3.4.6, which is applicable in Hot Operating Modes (Reference 1).

CNS Basis Reference(s):

1. Technical Specification LCO 3.4.6.

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 > 30 gpm (NOTE 6)

NOTE 6 – See Table F-1, Fission Product Barrier Matrix, for possible escalation above the Unusual Event due to RCS Leakage.

Mode Applicability:

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

NEI 99-01 Basis:

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

(continued on next page)

CNS Basis:

- Leakage is monitored by utilizing the following techniques (Reference 1):
- Sensing excess flow in piping systems.
- Sensing pressure and temperature changes in the Primary Containment.
- Monitoring for high flow and temperature through selected drains.
- Sampling airborne particulate and gaseous radioactivity.
- Drywell floor and equipment drain sump leak rate alarm system.

The 10 gpm value for the unidentified drywell leakage was selected because it is observable with normal Control Room measurement of sump pump out rates (e.g., Drywell Sump Pump Flow RW-FR-528, red/blue pen, etc.). Drywell equipment Sump G and drywell floor drain Sump F each have a FILL UP RATE HIGH annunciator on Panel 9-4-2. If either sump fills from the low-level switch reset point to the high-level pump start point before a preset timer has timed out, the annunciator will alarm indicating the sumps are filling at an excessive rate. Sumps F and G will overflow to each other through a trench system. Drywell equipment and floor drain sump pump isolation valves isolate on RPV low water level (≥ 3 in.) or high drywell pressure (≤ 1.84 psig).

A SRV that opens but cannot be closed from the Control Room meets this criterion and the UE should be declared.

CNS Basis Reference(s):

1. System Operating Procedure 2.2.27, Equipment, Floor, and Chemical Drain System.
2. Alarm Procedure 2.3_9-4-2, Panel 9-4 - Annunciator 9-4-2, B-1, DRYWELL EQUIP SUMP G HIGH FILL-UP RATE.
3. Alarm Procedure 2.3_9-4-2, Panel 9-4 - Annunciator 9-4-2, B-2, DRYWELL FLOOR DRN SUMP F HI FILL-UP RATE.
4. Surveillance Procedure 6.LOG.601, Daily Surveillance Log - Modes 1, 2, and 3.
5. Technical Specification LCO 3.4.4, RCS Operational Leakage.
6. Technical Specification LCO 3.4.5, RCS Leakage Detection Instrumentation.
7. USAR Section X-14.0, Equipment and Floor Drainage Systems.

Category: S - System Malfunction

Subcategory: 7 - Loss of DC Power

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

EAL:

SS7.1 Site Area Emergency

< 105 VDC bus voltage indications on **all** vital 125 VDC buses (1A and 1B) 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 - Power Operation, 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 Abnormal Rad Levels/Radiological Effluent, Fission Product Barrier Degradation.

CNS Basis:

105 VDC is the minimum design bus voltage (Reference 4).

The 125 VDC System supplies DC power to conventional station emergency equipment and selected Safeguard System loads. 125 VDC Distribution Panels supply control and instrument power for annunciators control logic power, and protective relaying. Figure S-2 illustrates the 125 VDC Power System (Reference 3).

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If 125 VDC Distribution Panel A is lost, the following major equipment is affected: RRMG A speed and breaker control, 4160V Bus 1A, 1E, and 1F breaker control and undervoltage logics, 480V Bus 1A and 1F breaker control, the right light in all Control Room annunciators, annunciator panels for Water Treatment, RHR A Gland Water, Auxiliary Steam Boiler C, DG-1 starting and breaker control logics, CS A, RCIC, and RHR A control logics, TIP valve control monitors, main generator voltage regulation, RFPT A trip logic, and ARI solenoid valve power.

If 125 VDC Distribution Panel B is lost, the following major equipment is affected: RRMG B speed and breaker control, 4160V Bus 1B and 1G breaker control and undervoltage logics, 480V Bus 1B and 1G breaker control, the left light in all Control Room annunciators, annunciator panels for ALRW, RHR B Gland Water, Auxiliary Steam Boiler D, DG-2 starting and breaker control logics, CS B, HPCI, and RHR B control logics, main generator trip logic, main generator and transformer protective relaying, bypass valves fail to control pressure after turbine trip and RFPT B trip logic.

Battery chargers receive their power from 460V critical motor control centers. Each 125 VDC bus receives power from either a 125 VDC battery or a 125 VDC battery charger. The battery chargers receive their power from 460V critical motor control centers. The 250 VDC System supplies DC power to conventional station emergency equipment and selected Safeguard System loads. Although 250 VDC Buses 1A and 1B provide vital DC emergency power, 250 VDC Safety System loads (such as motor operated valves) also require 125 VDC control power. Loss of 125 VDC buses alone, therefore, would render most Safeguard System loads inoperable (Reference 4, 5, 6).

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

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CNS Basis Reference(s):

1. Emergency Procedure 5.3DC125, Loss of 125 VDC.
2. Surveillance Procedure 6.EE.607, 125V Station Battery Modified Performance Discharge Test.
3. BR 3058 DC One Line Diagram.
4. Technical Specifications B 3.8.4.
5. USAR Section VIII-6.2.
6. USAR Section VIII-6.3.

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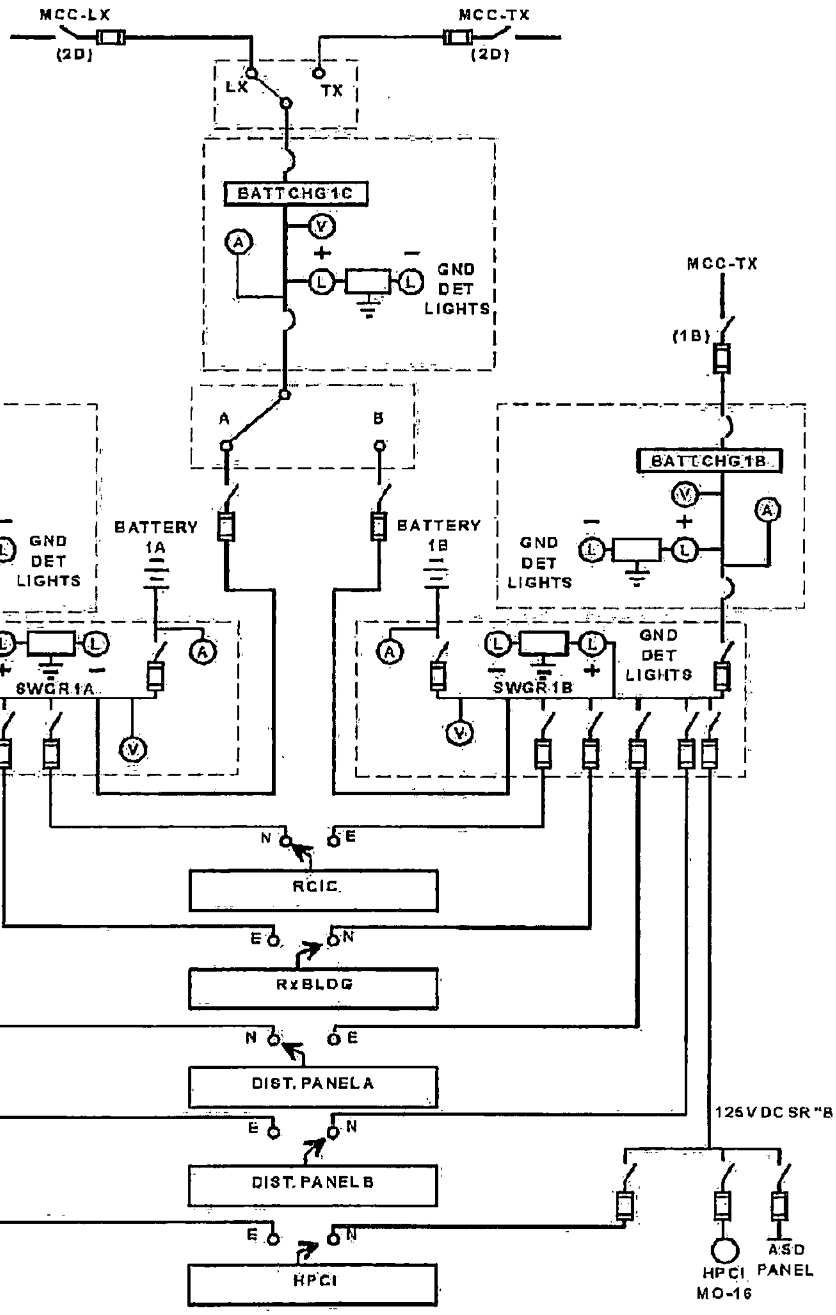
Figure S-2: 125 VDC Power System



Category: S - System Malfunction

Subcategory: 8 - Communications

Initiating Condition: Loss of **all** on-site or off-site communications capabilities

EAL:

SU8.1 Unusual Event

Loss of **all** Table S-2 on-site (internal) communications capability affecting the ability to perform routine operations

OR

Loss of **all** Table S-2 off-site (external) communications methods affecting the ability to perform off-site notifications

Table S-2 Communications Systems

System	Onsite (internal)	Offsite (external)
Station Intercom System "Gaitronics"	X	
Site UHF Radio Consoles	X	
Radio Paging System	X	
Alternate Intercom	X	
Sound Power System	X	
CNS On-Site Cell Phone System	X	X
Telephone system (PBX)	X	X
Federal Telecommunications System (FTS 2001)		X
Local Telephones (C.O. Lines)		X
CNS State Notification Telephones		X
Satellite Telephones		X

Mode Applicability:

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

(continued on next page)

NEI 99-01 Basis:

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

The availability of one method of ordinary off-site 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 radio transmissions, individuals being sent to off-site locations, etc.) are being utilized to make communications possible.

CNS Basis:

NOTE – EPIP 5.7COMMUN has more detail on each of the communications systems covered by this EAL.

On-site/off-site communications include one or more of the systems listed in Table S-2 (Reference 1).

- Station Intercom System "Gaitronics": Permits communication between the different parts of the plant and it also incorporates a public address system for plant wide announcements.
- Site UHF Radio Consoles: The site UHF radio system uses four repeaters; Base 1 and Base 2 are used by Operations, Base 3 and Base 4 are used by Security. These repeaters operate on different frequencies. All remote control, portable, and mobile units are capable of selecting either repeater.
- Radio Paging System: CNS leases pagers and radio paging services from a telecommunications company. Pagers are issued to various Management and Emergency Response personnel at CNS and other NPPD locations. Pagers can be activated from any touch-tone phone, on-site or off-site.
- Alternate Intercom: Provides an alternate in-plant communications network utilizing the back-up tone commander PBX System. This system is located in the ERP shack and has battery back-up.
- Sound Powered System.
- CNS On-Site Cell Phone System.

(continued on next page)

- Telephone System (PBX): Provides voice communication between virtually all buildings, offices, and operation facilities within the station. The telephone system also provides communications between the plant and off-site facilities via the telephone switchboard network. The system allows Operating Crews to alert plant personnel in emergencies. The telephone company provides the normal and leased line services.
- Federal Telecommunications System (FTS 2001): The Health Physics Network (HPN) and Emergency Notification System (ENS) provide communications between NRC and CNS during an emergency.
- Local Telephones (C.O. Lines).
- CNS State Notification Telephones: The CNS State Notification Telephone System is the primary means for the plant to make emergency notifications to state and local authorities. This system provides direct communication with the Nebraska State Patrol, the Missouri State Patrol, the Atchison County Sheriff's Department, and the Nemaha and Richardson County Sheriff's Departments.
- Satellite Telephones.

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

CNS Basis Reference(s):

1. EPIP 5.7COMMUN, Communications, Emergency Response Facility Communication Equipment attachment.

Category: E - ISFSI

Subcategory: None

Initiating Condition: Damage to a loaded cask confinement boundary

EAL:

EU1.1 Unusual Event
Damage to a loaded cask confinement boundary

Mode Applicability:

N/A

NEI 99-01 Basis:

An Unusual Event in this EAL 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.

CNS Basis:

Minor surface damage that does not affect storage cask boundary is excluded from the scope of this EAL.

CNS Basis Reference(s):

1. Certificate of Compliance Number 1004 Amendment 9, April 17, 2007.

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)

Mode Applicability:

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

NEI 99-01 Basis:

None

CNS Basis:

Fuel Clad, RCS, and Primary Containment comprise the fission product barriers. Table F-1 (Attachment 3) 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.

CNS Basis Reference(s):

None

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)

Mode Applicability:

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

NEI 99-01 Basis:

None

CNS Basis:

Fuel Clad, RCS, and Primary Containment comprise the fission product barriers. Table F-1 (Attachment 3) 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.

CNS Basis Reference(s):

None

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)

Mode Applicability:

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

NEI 99-01 Basis:

None

CNS Basis:

Fuel Clad, RCS, and Primary Containment comprise the fission product barriers. Table F-1 (Attachment 3) 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).

(continued on next page)

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 off-site 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.

CNS Basis Reference(s):

None

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)

Mode Applicability:

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

NEI 99-01 Basis:

None

CNS Basis:

Fuel Clad, RCS, and Primary Containment comprise the fission product barriers. Table F-1 (Attachment 3) 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.

CNS Basis Reference(s):

None

ATTACHMENT 3 FISSION PRODUCT BARRIERS - INDICATIONS OF LOSS OR POTENTIAL LOSS TECHNICAL BASES (INFORMATION USE)

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 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. ERD.

E. Rad.

F. 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 so that they can be easily identified.

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.

(continued on next page)

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 if 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 if 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 according category Loss followed by category Potential Loss beginning with Category A, then B, then C, then D, then E, and then F.

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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. PC flooding is required due to any of the following: • RPV water level cannot be restored and maintained > -183 in. • RPV water level cannot be restored and maintained ≥ -209 in. and no core spray subsystem flow can be restored and maintained ≥ 4,750 gpm • RPV water level cannot be determined and core damage is occurring	8. RPV level cannot be restored and maintained > -158 in. or cannot be determined	10. RPV level cannot be restored and maintained > -158 in. or cannot be determined	None	None	25 PC Flooding required
B. PC Pressure / Temperature	None	None	11. PC pressure > 1.84 psig due to RCS leakage	None	19. PC pressure rise followed by a rapid unexplained drop in PC pressure 20. PC pressure response not consistent with LOCA conditions	26. PC pressure > 56 psig and rising 27. Deflagration concentrations exist inside PC ≥ 6% H_2 in drywell or torus (or cannot be determined) AND ≥ 5% O_2 in drywell or torus (or cannot be determined) 28. Average torus water temperature and RPV pressure cannot be maintained within the Heat Capacity Temperature Limit (EOP/SAG Graph 7)
C. Isolation	None	None	12. 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	16. RCS leakage > 50 gpm inside the drywell 17. Unisolable primary system discharge outside primary containment as indicated by exceeding any secondary containment Maximum Normal Operating temperature or radiation value (EOP-5A Tables 9 and 10)	21. Failure of all valves in any one line to close AND Direct downstream pathway to the environment exists after PC isolation signal 22. Intentional PC venting per EOPs 23. Unisolable primary system discharge outside PC as indicated by exceeding any secondary containment Maximum Safe Operating temperature or radiation value (EOP-5A Tables 9 and 10)	None
D. ERD	None	None	13. Emergency RPV depressurization is required	None	None	None
E. Rad	2. Drywell radiation monitor (RMA-RM-40A/B) > 2.50E+03 Rem/hr 3. Primary coolant activity > 300 μCi/gm dose equivalent I-131 4. Main Steam Line Radiation Monitor Readings ≥ Hi-Hi Alarm Setpoint 5. ≥ 1.5E4 mrem/hr on SJAE monitor 6. Non-LOCA with DW Rad Monitor reading > 115 REM/hr	None	14. Drywell radiation monitor (RMA-RM-40A/B) > 2.40E+02 Rem/hr	None	None	29. Drywell radiation monitor (RMA-RM-40A/B) > 5.00E+04 Rem/hr
F. Judgment	7. Any condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier.	9. Any condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier	15. Any condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	18. Any condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	24. Any condition in the opinion of the Emergency Director that indicates loss of the PC barrier	30. Any condition in the opinion of the Emergency Director that indicates potential loss of the PC barrier

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Loss

Threshold:

1. Primary Containment flooding is required due to **any** of the following:
   - RPV water level **cannot** be restored and maintained > -183 in.
   - RPV water level **cannot** be restored and maintained ≥ -209 in. and **no** core spray subsystem flow can be restored and maintained ≥ 4,750 gpm
   - RPV water level **cannot** be determined and core damage is occurring

NEI 99-01 Basis:

The "Loss" threshold 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.

CNS Basis:

EOP-1A, EOP-2B, EOP-7A, and EOP-7B specify entry to the SAGs when core cooling is severely challenged and Primary Containment flooding is required. SAG entry signifies the need to flood the Primary Containment. 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. Primary Containment flooding and SAG entry is required when any of the following conditions exist (Reference 1):

- RPV water level cannot be restored and maintained above -183 in. (MSCRWL, EOP-1A/7A) (Reference 2, 3, 6).
- RPV water level cannot be restored and maintained at or above -209 in. (elevation of the jet pump suction) and no core spray subsystem flow can be restored and maintained equal to or ≥ 4,750 gpm (design core spray flow, EOP-1A) (Reference 2, 7).
- RPV water level cannot be determined and core damage is occurring (EOP-2B/7B) (Reference 4, 5).

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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 22). Since the EOP requirement for Primary Containment flooding is reached after core uncovery has occurred a Loss of the RCS barrier exists (RCS Loss 7). Primary Containment flooding and SAG entry, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

CNS Basis Reference(s):

1. AMP-TBD00 PSTG/SATG Technical Bases, Contingency #1, #4, #5.
2. EOP-1A, RPV Control.
3. EOP-7A, RPV Level (Failure-to-Scram).
4. EOP-2B, RPV Flooding.
5. EOP-7B, RPV Flooding (Failure-to-Scram).
6. NEDC 97-090J.
7. NEDC 97-089.

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

8. RPV level cannot be restored and maintained > -158 in. or cannot be determined

NEI 99-01 Basis:

This threshold is the same as the RCS barrier "Loss" Threshold A.10 and corresponds to the 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.

CNS Basis:

The KEY consideration of this criterion is adequate core cooling. RPV conditions that result in inadequate core cooling shall result in determinations that the fuel clad barrier has been potentially lost and the RCS barrier has been lost. A RPV level instrument reading of -158 in. indicates RPV level is at the top of active fuel (TAF) (Reference 4). 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). 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 after RPV pressure reduction (either manually, automatically, or by failure of the RCS barrier), RPV level cannot be restored and maintained above TAF, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the decreasing level trend. 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 below TAF, the level is indicative of a challenge to core cooling and the Fuel Clad barrier.

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EOPs allow the Operator a wide choice of RPV injection sources to consider when evaluating if RPV water level can be restored and maintained to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, an important consideration is the status of adequate core cooling during the pressure reduction process. A rapid depressurization which allows refilling the RPV with a high volume system will normally maintain adequate cooling with steam flow. Slow depressurizations, however, may challenge adequate core cooling due to both low water levels and low steam flows.

A threshold prescribing declaration when a parameter value cannot be maintained above a specified limit does not require immediate action simply because the current value is below the limit (top of active fuel), but **does not permit extended operation below the limit.** Because the systems used to restore water level have different volume and discharge pressure capabilities, and pressure control methods also have varying rates that RPV pressure can be modified to support injection, the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained within a reasonable amount of time. Determination of inability to restore and maintain RPV level is based on actions driven by EOPs to restore level.

In high-power ATWS/failure to scram events, EOPs may direct the Operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs SA2, SS2, and SG2 will dictate an emergency classification. The fission product barrier criteria should continue to be evaluated independently to identify barrier conditions that would require escalation of the classification.

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When RPV level cannot be determined, EOPs require entry to EOP-2B, RPV Flooding, or EOP-7B, RPV Flooding (Failure-to-Scram). RPV water level indication provides the primary means of knowing if 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-2B/7B 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 (Reference 1, 3).

CNS Basis Reference(s):

1. EOP-2B, RPV Flooding.
2. EOP-7A, RPV Level (Failure-to-Scram).
3. EOP-7B, RPV Flooding (Failure-to-Scram).
4. NEDC 97-089.

Barrier: Fuel Clad

Category: E. Rad

Degradation Threat: Loss

Threshold:

2. Drywell radiation monitor (RMA-RM-40A/B) > 2.50E+03 Rem/hr

NEI 99-01 Basis:

2.50E+03 Rem/hr is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the drywell.

The reading was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of approximately 300 μCi/gm dose equivalent I-131 (~ 1% clad damage) into the drywell atmosphere.

Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within Technical Specifications and are therefore indicative of fuel damage.

This value is higher than that specified for RCS barrier Loss Threshold 14. 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.

CNS Basis:

EPIP 5.7.17.1, Dose Assessment (Manual), Core Damage Estimation attachment provides a method of calculating percent fuel clad damage and fuel melt based on drywell radiation. Under LOCA conditions, a reading of 2.44E+6 Rem/hr corresponds to 100% core melt on drywell radiation Monitors RMA-RM-40A/B. A value of 2.44E+3 Rem/hr (rounded to 2.50E+03 Rem/hr) yields 1% fuel clad damage using this method.

In order to reach this Fuel Clad barrier Potential Loss threshold, a loss of the RCS barrier has already occurred (see RCS Loss 14). This threshold, therefore, represents at least a Site Area Emergency classification.

CNS Basis Reference(s):

1. EPIP 5.7.17.1, Dose Assessment (Manual).

Barrier: Fuel Clad

Category: E. Rad

Degradation Threat: Loss

Threshold:

3. Primary coolant activity > 300 μCi/gm dose equivalent I-131

NEI 99-01 Basis:

Coolant activity of 300 μCi/gm dose equivalent I-131 is well above that expected for iodine spikes and corresponds to about 1% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus, the Fuel Clad Barrier is considered lost.

CNS Basis:

None

CNS Basis Reference(s):

1. EPIP 5.7.17, CNS-DOSE Assessment, and/or EPIP 5.7.17.1, Dose Assessment (Manual).
2. NEI 99-01, Revision 5.
3. NEDC 02-20, Estimation of Reactor Coolant System Dose Equivalent I-131 Concentration Following a 1% Fuel Clad Failure (Degraded Core) Under Non-LOCA Conditions.

Barrier: Fuel Clad

Category: E. Rad

Degradation Threat: Loss

Threshold:

4. Main Steam Line Radiation Monitor Readings ≥ Hi-Hi Alarm Setpoint

NEI 99-01 Basis:

None

CNS Basis:

The Hi-Hi alarm setpoint for the Main Steam Line Radiation Monitors is based on a Control Rod Drop Accident. This accident is most severe when initiated at < 10% rated thermal power. The setpoint is a multiple of the normal full power background reading on these monitors that was observed during the previous operating cycle. Clad damage resulting in the Hi-Hi Main Steam Line Radiation Monitor Alarms due to a Control Rod Drop Accident could exceed 2% of the core's fuel rods. USAR XIV-6.2 discusses the Control Rod Drop Accident. Specific Emergency Director evaluation is required to determine if the requisite plant conditions exist that would make this criteria valid.©[3]

CNS Basis Reference(s):

1. EPIP 5.7.17, CNS-DOSE Assessment, and/or EPIP 5.7.17.1, Dose Assessment (Manual).
2. USAR XIV-6.2, Control Rod Drop Accident.

Barrier: Fuel Clad

Category: E. Rad

Degradation Threat: Loss

Threshold:

5. ≥ 1.5E4 mrem/hr on SJAE Monitor

NEI 99-01 Basis:

None

CNS Basis:

NEDC 02-004 assumes that cladding damage has resulted in 300 μCi/gm of I-131 in the reactor coolant, that non-LOCA conditions exist, and that normal sample flow through the monitor is maintained. Under these conditions, the SJAE monitors are expected to reach 15,000 mrem/hr. If sample flow is lost or the monitors are isolated and they reach this point, they are assumed to have failed. Specific Emergency Director evaluation is required to determine if the requisite plant conditions exist that would make this criteria valid.©[3]

CNS Basis Reference(s):

1. EPIP 5.7.17, CNS-DOSE Assessment, and/or EPIP 5.7.17.1, Dose Assessment (Manual).
2. NEDC 02-004, Estimation of the Steam Jet Air Ejector Radiation Monitor RMP-RM-150A(B), Readings Following a 1% Fuel Clad Release (Degraded Core) in the Reactor Coolant System.

Barrier: Fuel Clad

Category: E. Rad

Degradation Threat: Loss

Threshold:

6. Non-LOCA with DW Rad Monitor reading > 115 REM/hr

NEI 99-01 Basis:

None

CNS Basis:

NEDC 02-009 assumes that cladding damage has resulted in 300 μCi/gm of I-131 in the reactor coolant and that non-LOCA conditions exist. The calculated value is based strictly on shine from the RR System piping, and as a result, is only valid if primary containment parameters do not indicate increased RCS leakage. With increased RCS leakage, any contaminants in the coolant could be in closer proximity to the detectors (as an aerosol or vapor) causing the monitors to read higher. If elevated RCS leakage is suspected, a drywell radiation monitor reading of 2500 R/hr, per Criteria 2, is the appropriate threshold for this parameter to determine Fuel Clad failure. Specific Emergency Director evaluation is required to determine if the requisite plant conditions exist that would make this criteria valid.©[3]

CNS Basis Reference(s):

1. EPIP 5.7.17, CNS-DOSE Assessment, and/or EPIP 5.7.17.1, Dose Assessment (Manual).
2. NEDC 02-009, Estimation of Primary Containment High Range Monitor RMA RM 40A(B), Readings Following 1% Clad Failure in the RCS Under Non-LOCA Conditions.

Barrier: Fuel Clad

Category: F. Judgment

Degradation Threat: Loss

Threshold:

7. **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 considered as a factor in Emergency Director judgment that the barrier may be considered lost.

CNS Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining if 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 off-site 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.

CNS Basis Reference(s):

None

Barrier: Fuel Clad

Category: F. Judgment

Degradation Threat: Potential Loss

Threshold:

9. **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 considered as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

CNS Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Fuel Clad 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 off-site 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.

CNS Basis Reference(s):

None

Barrier: Reactor Coolant System

Category: A. RPV Level

Degradation Threat: Loss

Threshold:

10. RPV level **cannot** be restored and maintained > -158 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 Fuel Clad Barrier Potential Loss Threshold #8 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.

CNS Basis:

The KEY consideration of this criterion is the ability to maintain adequate core cooling. RPV conditions that result in inadequate core cooling shall result in determinations that the fuel clad barrier has been potentially lost and the RCS barrier has been lost. A RPV level instrument reading of -158 in. indicates RPV level is at the top of active fuel (TAF) (Reference 4). 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 after RPV pressure reduction (either manually, automatically, or by failure of the RCS barrier), RPV level cannot be restored and 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.

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EOPs allow the Operator a wide choice of RPV injection sources to consider when evaluating if RPV water level can be restored and maintained to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, an important consideration is the status of adequate core cooling during the pressure reduction process. A rapid depressurization which allows refilling the RPV with a high volume system will normally maintain adequate cooling with steam flow. Slow depressurizations, however, may challenge adequate core cooling due to both low water levels and low steam flows.

A threshold prescribing declaration when a parameters value cannot be maintained above a specified limit does not require immediate action simply because the current value is below the limit (top of active fuel), **but does not permit extended operation below the limit.** Because the systems used to restore water level have different volume and discharge pressure capabilities, and pressure control methods also have varying rates that RPV pressure can be modified to support injection, the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained within a reasonable amount of time. Determination of inability to restore and maintain RPV level is based on actions driven by EOPs to restore level.

In high-power ATWS/failure to scram events, EOPs may direct the Operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs SA2, SS2, and SG2 will dictate an emergency classification. The fission product barrier criteria should continue to be evaluated independently to identify barrier conditions that would require escalation of the classification.

When RPV level cannot be determined, EOPs require entry to EOP-2B, RPV Flooding, or EOP-7B, RPV Flooding (Failure-to-Scram). The instructions in EOP-2B/7B specify emergency depressurization of the RPV, which is defined to be a Loss of the RCS barrier (RCS Loss 10) (Reference 1, 3).

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CNS Basis Reference(s):

1. EOP-2B, RPV Flooding.
2. EOP-7A, RPV Level (Failure-to-Scram).
3. EOP-7B, RPV Flooding (Failure-to-Scram).
4. NEDC 97-089.

Barrier: Reactor Coolant System

Category: B. PC Pressure/Temperature

Degradation Threat: Loss

Threshold:

11. PC pressure > 1.84 psig due to RCS leakage

NEI 99-01 Basis:

The threshold pressure value is the Primary Containment high pressure scram setpoint and is indicative of a LOCA event that requires ECCS response.

There is no Potential Loss threshold associated with this item.

CNS Basis:

The Primary Containment (PC) high pressure scram setpoint is an entry condition to EOP-1A, RPV Control, and EOP-3A, Primary Containment Control (Reference 1, 2). Normal Primary Containment pressure control functions (e.g., operation of drywell cooling, SBGT, etc.) are specified in EOP-3A in advance of less desirable but more effective functions (e.g., operation of drywell or torus sprays, etc.).

In the CNS design basis, Primary Containment pressures above the 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 (Reference 3).

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 > 1.84 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 > 1.84 psig should not be considered a RCS barrier Loss.

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CNS Basis Reference(s):

1. EOP-1A, RPV Control.
2. EOP-3A, Primary Containment Control.
3. USAR Section XIV-6.3.

Barrier: Reactor Coolant System

Category: C. Isolation

Degradation Threat: Loss

Threshold:

12. 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.

NEI 99-01 Basis:

An unisolable RCS break outside Primary Containment is a breach of the RCS barrier. Thus, this threshold is included for consistency with the Alert emergency classification level.

Large high-energy line breaks such as HPCI, Feedwater, RWCU, or RCIC that are unisolable represent a significant loss of the RCS barrier and should be considered as MSL breaks for purposes of classification.

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CNS 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 flowpath 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 21) 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.

CNS Basis Reference(s):

1. System Operating Procedure 2.2.33, High Pressure Coolant Injection System.
2. System Operating Procedure 2.2.56, Main Steam System.
3. System Operating Procedure 2.2.66, Reactor Water Cleanup.
4. System Operating Procedure 2.2.67, Reactor Core Isolation Cooling System.
5. BR 2041, Reactor Building Main Steam System.
6. BR 2042.
7. BR 2043.
8. BR 2044, HPCI System.

Barrier: Reactor Coolant System

Category: C. Isolation

Degradation Threat: Potential Loss

Threshold:

16. 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.

CNS 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 (Reference 1-6). 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.

CNS Basis Reference(s):

1. System Operating Procedure 2.2.27, Equipment, Floor, and Chemical Drain System.
2. Alarm Procedure 2.3_9-4-2, Panel 9-4 - Annunciator 9-4-2, B-1/B-2.
3. Surveillance Procedure 6.LOG.601, Daily Surveillance Log - Modes 1, 2, and 3.
4. Technical Specifications LCO 3.4.4, RCS Operational Leakage.
5. Technical Specifications LCO 3.4.5, RCS Leakage Detection Instrumentation.
6. USAR Section X-14.0, Equipment and Floor Drainage Systems.

Barrier: Reactor Coolant System

Category: C. Isolation

Degradation Threat: Potential Loss

Threshold:

17. Unisolable primary system discharge outside Primary Containment as indicated by exceeding **any** Secondary Containment Maximum Normal Operating temperature or radiation value (EOP-5A Tables 9 and 10)

NEI 99-01 Basis:

Potential loss of RCS based on primary system leakage outside the Primary Containment is determined from temperature or area radiation Maximum Normal Operating values (EOP-5A, Tables 9 and 10) 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 20 (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

CNS 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 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-5A, Secondary Containment Control, Tables 9 and 10 (Reference 1) (see below).

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In general, multiple indications should be used to determine if 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 flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the Secondary Containment. As indicated by Note 5 in EOP-5A, Table 10, RP surveys and ARM teledosimetry system may be used for these indications.

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EOP-5A Table 9 - Secondary Containment Temperatures

9	SECONDARY CONTAINMENT TEMPERATURES SPDS 16			
Maximum Normal Operating Value		Maximum Safe Operating Value		
Area	Any Temp. Switch Alarmed	Area	Value (°F)	Actual Value
NE Quad	RCIC-TS-77A RCIC-TS-77C	NE Quad	195	
SE Quad	RWCU-TS-117F	SE Quad	195	
NW Quad	RHR-TS-99C	NW Quad	195	
SW Quad and HPCI Room	RHR-TS-99G HPCI-TS-105B HPCI-TS-105D	SW Quad and HPCI Room	195	
1001' El. 976' El. 958' El.	RWCU-TS-117B	1001' El. 976' El. 958' El.	195	
903' El. and 931' El.	RHR-TS-99A RHR-TS-99E MS-TS-126A MS-TS-126C RWCU-TS-117E RWCU-TS-117A HPCI-TS-105A	903' El. and 931' El.	195	

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EOP-5A Table 10 - Secondary Containment Radiation Levels

10 SECONDARY CONTAINMENT RADIATION LEVELS (5)
SPDS 15

Maximum Normal Operating Value			Maximum Safe Operating Value		
Area	Any ARM Alarmed	Range (mR/hr)	Area	Value (mR/hr)	Actual Value
FUEL POOL AREA	RMA-RA-1	100 - 10^6	1001' El.	1000	
FUEL POOL AREA	RMA-RA-2	.01 - 100	1001' El.		
RWCU PRECOAT AREA	RMA-RA-4	0.1 - 1000	958' El.		
RWCU SLUDGE AND DECANT PUMP AREA	RMA-RA-5	0.1 - 1000	931' El.	1000	
CRD HYDRAULIC EQUIP AREA (SOUTH)	RMA-RA-8	.01 - 100	903' El.		
CRD HYDRAULIC EQUIP AREA (NORTH)	RMA-RA-9	.01 - 100			
HPCI PUMP ROOM	RMA-RA-10	.01 - 100	HPCI Room		
RHR PUMP ROOM, (SOUTHWEST)	RMA-RA-11	.01 - 100	SW Quad	1000	
TORUS HPV AREA (SOUTHWEST)	RMA-RA-27	1.0 - 10000	SW Torus		
RHR PUMP ROOM, (NORTHWEST)	RMA-RA-12	.01 - 100	NW Quad	1000	
RCIC/CORE SPRAY PUMP ROOM, (NORTHEAST)	RMA-RA-13	.01 - 100	NE Quad	1000	
CORE SPRAY PUMP ROOM, (SOUTHEAST)	RMA-RA-14	.01 - 100	SE Quad	1000	

NOTE 5

Area radiation levels can be monitored by RP surveys or ARM teledosimetry system

CNS Basis Reference(s):

1. EOP-5A, Secondary Containment Control.

Barrier: Reactor Coolant System

Category: D. ERD

Degradation Threat: Loss

Threshold:

13. Emergency RPV Depressurization is required

NEI 99-01 Basis:

Plant symptoms requiring Emergency RPV Depressurization are specified in the EOPs (Reference 1, 2, 3, 4, 5) and 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 regardless of any subsequent radiological release rate (Reference 1, 5). 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.

CNS Basis:

None

CNS Basis Reference(s):

1. EOP-1A, RPV Control.
2. EOP-2A, Steam Cooling.
3. EOP-3A, Primary Containment Control.
4. EOP-5A, Secondary Containment Control, Radioactivity Release Control.
5. EOP-7A, RPV Control (Failure-to-Scram).

Barrier: Reactor Coolant System

Category: E. Rad

Degradation Threat: Loss

Threshold:

14. Drywell radiation monitor (RMA-RM-40A/B) > 2.40E+02 Rem/hr - LOCA

NEI 99-01 Basis:

The 2.40E+02 Rem/hr value indicates the release of reactor coolant to the Primary Containment.

The reading was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the drywell atmosphere.

This reading is less than that specified for Fuel Clad barrier Loss Threshold 2. Thus, this threshold would be indicative of a 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.

CNS Basis:

EPIP 5.7.17.1, Dose Assessment (Manual), Core Damage Estimation attachment provides a method of calculating percent fuel clad damage and fuel melt based on drywell radiation. Under LOCA conditions, a reading of 2.44E+6 Rem/hr corresponds to 100% core melt on RMA-RM-40A/B. A value of 2.44E+2 Rem/hr (rounded to 2.40E+02 Rem/hr) yields 0.1% fuel clad damage using this method. This amount of clad damage is approximately the equivalent of Technical Specification coolant activity discharged uniformly throughout the Primary Containment (Reference 1).

CNS Basis Reference(s):

1. EPIP 5.7.17.1, Dose Assessment (Manual).

Barrier: Reactor Coolant System

Category: F. Judgment

Degradation Threat: Loss

Threshold:

15. **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.

CNS Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining if 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 off-site 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.

CNS Basis Reference(s):

None

Barrier: Reactor Coolant System

Category: F. Judgment

Degradation Threat: Potential Loss

Threshold:

18. **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.

CNS Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining if 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 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 off-site 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.

CNS Basis Reference(s):

None

Barrier: Primary Containment

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

25. PC Flooding required

NEI 99-01 Basis:

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 the SAGs (Primary Containment Flooding procedures) is a logical escalation in response to the inability to maintain adequate core cooling.

SAGs direct the Operators to perform Containment Flooding when Reactor Vessel Level cannot be restored and maintained greater than specified values or RPV level cannot be determined with indication that core damage is occurring.

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.

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CNS Basis:

EOP-1A, EOP-2B, EOP-7A, and EOP-7B specify entry to the SAGs when core cooling is severely challenged. SAG entry signifies the need to flood the Primary Containment. 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 (Reference 1):

- RPV water level cannot be restored and maintained above -183 in. (MSCRWL) (Reference 5).
- RPV water level cannot be restored and maintained at or above -209 in. (elevation of the jet pump suction) and no core spray subsystem flow can be restored and maintained ≥ 4,750 gpm (design core spray flow) (Reference 4).
- RPV water level cannot be determined and core damage is occurring (Reference 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 1). Since SAG entry occurs after core uncovery has occurred, a Loss of the RCS barrier exists (RCS Loss 7). SAG entry, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

CNS Basis Reference(s):

1. AMP-TBD00 PSTG/SATG Technical Bases, Contingency #1, #4, #5.
2. EOP-2B, RPV Flooding.
3. EOP-7B, RPV Flooding (Failure-to-Scram).
4. NEDC 97-089.
5. NEDC 97-090J.

Barrier: Primary Containment

Category: B. PC Pressure/Temperature

Degradation Threat: Loss

Threshold:

19. 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.

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.

CNS Basis:

None

CNS Basis Reference(s):

None

Barrier: Containment

Category: B. PC Pressure/Temperature

Degradation Threat: Loss

Threshold:

20. PC pressure response **not** consistent with LOCA conditions

NEI 99-01 Basis:

Primary Containment pressure should increase initially as a result of mass and energy release into containment from a LOCA. Thus, Primary Containment pressure not initially 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.

CNS Basis:

Analysis of the Primary Containment response to a postulated DBA LOCA event gives a peak drywell pressure of 54.4 psig and a peak drywell temperature of 301.4°F. These peak values were obtained for the power/flow point of 102%P/75%F (MELLL point). Due to conservatisms in LOCA analyses, actual pressure response is expected to be less than the analyzed response. For example, blowdown mass flowrate may be only 60% to 80% of the analyzed rate. The unexpected response is important because it is the indicator for a containment bypass condition (Reference 1).

As stated in the NEI 99-01 basis, the anticipated response to a LOCA is that the Primary Containment pressure would increase. The failure of torus to drywell vacuum breaker(s) could cause peak Primary Containment pressure to be higher than the analyzed peak Primary Containment pressure, but this condition is addressed by the potential containment failure Criteria 26. As such, violation of the pressure suppression pressure curve (PSP) does not constitute a loss of the Primary Containment.

CNS Basis Reference(s):

1. USAR Section XIV-6.3.7.

Barrier: Primary Containment

Category: B. PC Pressure/Temperature

Degradation Threat: Potential Loss

Threshold:

26. PC pressure > 56 psig and rising

NEI 99-01 Basis:

The 56 psig for Potential Loss of containment is based on the Primary Containment design pressure.

CNS Basis:

The Primary Containment internal design pressure is 56 psig (Reference 1). If this threshold is exceeded, a challenge to the Primary 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.

CNS Basis Reference(s):

1. USAR Table V-2-1.

Barrier: Primary Containment

Category: B. PC Pressure/Temperature

Degradation Threat: Potential Loss

Threshold:

27. Deflagration concentrations exist inside PC
    - $\geq$ 6% H_2 in drywell or torus (or **cannot** be determined)

    AND

    - $\geq$ 5% O_2 in drywell or torus or **cannot** be determined)

NEI 99-01 Basis:

BWRs specifically define the limits associated with explosive (deflagration) mixtures in terms of deflagration concentrations of hydrogen and oxygen. For Mk I/II containments, the deflagration limits are "6% hydrogen and 5% oxygen in the drywell or suppression chamber".

CNS 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.

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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) and readily recognizable because 6% hydrogen is well above the EOP-3, Primary Containment Control, entry condition (Reference 2, 3). Since the EOPs/SAGs require deflagration concentration actions to be performed when hydrogen and oxygen concentrations cannot be determined, the phrase has been added to the meaning of explosive mixtures. The minimum global deflagration hydrogen/oxygen concentrations (6% and 5%, respectively) require intentional Primary Containment venting, which is defined to be a Loss of Containment (PC Loss 22).

Drywell and suppression chamber atmosphere is monitored for H_2 and O_2 by a divisionally separated H_2/O_2 Monitoring System. The system consists of two H_2/O_2 analyzers (PC-AN-H2/O2I and PC-AN-H2/O2II), two remote process panels (PC-CS-H2/O2I and PC-CS-H2/O2II), two H_2 recorders (PC-R-H2I and PC-R-H2II), two O_2 recorders (PC-R-O2I and PC-R-O2II), an O_2 digital indicator (PC-I-1), associated control switches and sample stream indicating lights. H_2/O_2 analyzers are located in the Reactor Building at 976', remote process panels are located in the Cable Spreading Room, recorders are located on VBD-P1 and VBD-P2, the O_2 digital indicator and sample stream lights are located on VBD-H. Div 2 is normally in service providing O_2 concentration on VBD-H and H_2 and O_2 concentrations on PMIS (Reference 1).

CNS Basis Reference(s):

1. System Operating Procedure 2.2.60.1, Containment H2/O2 Monitoring System.

2. BWROG EPG/SAG Revision 2, Sections PC/G.

3. EOP-3A, Primary Containment Control.

Barrier: Primary Containment

Category: B. PC Pressure/Temperature

Degradation Threat: Potential Loss

Threshold:

28. Average torus water temperature and RPV pressure **cannot** be maintained within the Heat Capacity Temperature Limit (EOP/SAG Graph 7)

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.

CNS Basis:

This threshold is met when EOP-3, Primary Containment Control, Step SP/T-5 is reached (Reference 1).

CNS Basis Reference(s):

1. EOP-3A, Primary Containment Control.

Barrier: Primary Containment

Category: C. Isolation

Degradation Threat: Loss

Threshold:

21. Failure of **all** valves in **any** one line to close
AND
Direct downstream pathway to the environment exists after PC isolation signal

NEI 99-01 Basis:

These thresholds address 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% to 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.

CNS 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.

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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 non-isolable release path to the environment. These minor releases are assessed using the Category A, Abnormal Rad Release/Rad Effluent, EALs.

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. However, the expectation is that assessment, classification, and declaration of an emergency condition be made within 15 minutes after initial availability of the indication of a breach of Primary Containment integrity. 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-3A, Primary Containment Control, Step PC/P-6, may specify Primary Containment venting and intentional bypassing of the containment isolation valve logic, even if off-site radioactivity release rate limits are exceeded (Reference 1). Under these conditions with a valid containment isolation signal, the containment barrier should be considered lost under Criteria 22 for Intentional PC venting per EOPs.

CNS Basis Reference(s):

1. EOP-3A, Primary Containment Control.

Barrier: Primary Containment

Category: C. Isolation

Degradation Threat: Loss

Threshold:

22. 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 per 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.

CNS Basis:

EOP-3A, Primary Containment Control, Step PC/P-6, may specify Primary Containment venting and intentional bypassing of the containment isolation valve logic, even if off-site radioactivity release rate limits are exceeded (Reference 1). The threshold is met when the Operator begins venting the Primary Containment in accordance with EOP-3A, not when actions are taken to bypass interlocks prior to opening the vent valves. Purge and vent actions specified in EOP-3A, Step PC/P-1, to control Primary Containment pressure below the Primary Containment high pressure scram setpoint does not meet this threshold because such action is only permitted if off-site radioactivity release rates will remain below ODAM limits.

CNS Basis Reference(s):

1. EOP-3A, Primary Containment Control.

Barrier: Primary Containment

Category: C. Isolation

Degradation Threat: Loss

Threshold:

23. Unisolable primary system discharge outside PC as indicated by exceeding **any** Secondary Containment Maximum Safe Operating temperature or radiation value (EOP-5A, Tables 9 and 10)

NEI 99-01 Basis:

The presence of area radiation levels or area temperatures above any Maximum Safe Operating value indicates unisolable primary system leakage outside the Primary Containment are addressed 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.

CNS 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-5A, Secondary Containment Control, Tables 9 and 10 (see below).

In general, multiple indications should be used to determine if 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 flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the Secondary Containment. As indicated by NOTE 5 in EOP-5A, Table 10, RP surveys and ARM Teledosimetry System may be used for these indications.

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EOP-5A Table 9 - Secondary Containment Temperatures

9	SECONDARY CONTAINMENT TEMPERATURES SPDS 16			
Maximum Normal Operating Value		Maximum Safe Operating Value		
Area	Any Temp. Switch Alarmed	Area	Value (°F)	Actual Value
NE Quad	RCIC-TS-77A RCIC-TS-77C	NE Quad	195	
SE Quad	RWCU-TS-117F	SE Quad	195	
NW Quad	RHR-TS-99C	NW Quad	195	
SW Quad and HPCI Room	RHR-TS-99G HPCI-TS-105B HPCI-TS-105D	SW Quad and HPCI Room	195	
1001' El. 976' El. 958' El.	RWCU-TS-117B	1001' El. 976' El. 958' El.	195	
903' El. and 931' El.	RHR-TS-99A RHR-TS-99E MS-TS-126A MS-TS-126C RWCU-TS-117E RWCU-TS-117A HPCI-TS-105A	903' El. and 931' El.	195	

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EOP-5A Table 10 - Secondary Containment Radiation Levels

10	SECONDARY CONTAINMENT RADIATION LEVELS SPDS 15 (5)				
Maximum Normal Operating Value			Maximum Safe Operating Value		
Area	Any ARM Alarmed	Range (mR/hr)	Area	Value (mR/hr)	Actual Value
FUEL POOL AREA	RMA-RA-1	100 - 10^6	1001' El.	1000	
FUEL POOL AREA	RMA-RA-2	.01 - 100	1001' El.		
RWCU PRECOAT AREA	RMA-RA-4	0.1 - 1000	958' El.		
RWCU SLUDGE AND DECANT PUMP AREA	RMA-RA-5	0.1 - 1000	931' El.	1000	
CRD HYDRAULIC EQUIP AREA (SOUTH)	RMA-RA-8	.01 - 100	903' El.		
CRD HYDRAULIC EQUIP AREA (NORTH)	RMA-RA-9	.01 - 100			
HPCI PUMP ROOM	RMA-RA-10	.01 - 100	HPCI Room		
RHR PUMP ROOM, (SOUTHWEST)	RMA-RA-11	.01 - 100	SW Quad	1000	
TORUS HPV AREA (SOUTHWEST)	RMA-RA-27	1.0 - 10000	SW Torus		
RHR PUMP ROOM, (NORTHWEST)	RMA-RA-12	.01 - 100	NW Quad	1000	
RCIC/CORE SPRAY PUMP ROOM, (NORTHEAST)	RMA-RA-13	.01 - 100	NE Quad	1000	
CORE SPRAY PUMP ROOM, (SOUTHEAST)	RMA-RA-14	.01 - 100	SE Quad	1000	

NOTE 5

Area radiation levels can be monitored by RP surveys or ARM teledosimetry system

CNS Basis Reference(s):

1. EOP-5A, Secondary Containment Control.

Barrier: Primary Containment

Category: E. Rad

Degradation Threat: Potential Loss

Threshold:

29. Drywell radiation monitor (RMA-RM-40A/B) > 5.00E+04 Rem/hr

NEI 99-01 Basis:

50,000 Rem/hr is a value which indicates significant fuel damage well in excess of that required for loss of RCS and Fuel Clad. A major release of radioactivity requiring off-site 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 < 20%.

CNS Basis:

EPIP 5.7.17.1, Dose Assessment (Manual), Core Damage Estimation attachment provides a method of calculating percent fuel clad damage and fuel melt based on drywell radiation. A reading of 2.44E+6 Rem/hr corresponds to 100% core melt on RMA-RM-40A/B. A value of 4.88E+4 Rem/hr (rounded to 5.00E+04 Rem/hr) yields 20% fuel clad damage using this method (Reference 1).

CNS Basis Reference(s):

1. EPIP 5.7.17.1, Dose Assessment (Manual).

Barrier: Primary Containment

Category: F. Judgment

Degradation Threat: Loss

Threshold:

24. **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 Primary Containment barrier is lost. In addition, the inability to monitor the barrier should also be considered 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 Specification 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.

(continued on next page)

CNS Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining if 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 off-site 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.

CNS Basis Reference(s):

None

Barrier: Primary Containment

Degradation Threat: Potential Loss

Category: F. Judgment

Threshold:

30. **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 Primary Containment barrier is potentially lost. In addition, the inability to monitor the barrier should also be considered 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 Specification 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.

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CNS Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining if 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 off-site 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.

CNS Basis Reference(s):

None

ATTACHMENT 4 EAL CLASSIFICATION MATRIX (INFORMATION USE)

The following information defined in Attachment 1, EAL Scheme Explanation and Rationale, and contained in Attachment 2, Emergency Action Level Technical Bases, will be contained in the EAL Classification Matrix (Matrix or EAL Matrix):

- EAL Identifier.
- Mode Applicability.
- EAL.

The Matrix will also display the tables and notes from Attachments 2 and 3 applicable to the EALs. These items may be reformatted, arranged, and consolidated as required to facilitate use of the Matrix.

The EALs will be arranged by Emergency Class left to right, greatest to least, then by Category and subcategory top to bottom, and finally by EAL identifier top to bottom where required.

These Matrices will be controlled per this attachment. The information specified above will be word for word from Attachment 2 but may be formatted differently using different font sizes or color backgrounds to assist the visual presentation.

Each Matrix will contain a Revision data box that will list the current matrix revision number based on the information below:

EAL Classification Matrix Revision Data:

Procedure	EAL Classification Matrix Revision Number
EPIP 5.7.1, Attachment 4	Revision 13

It is not necessary that the Matrix revision number be revised with each revision of this procedure. However, if the Matrix is revised or if the information specified above (EALs, Notes, or Tables) are revised in Attachment 2, then Attachment 4 and the matrix must be revised to reflect the revised information.

Each controlled copy of the matrix will be labeled with the facility and copy number of the specific matrix card according to EPDG#2, Attachment F-5. Matrices that are not so labeled are uncontrolled and should be checked to verify the proper revision prior to use.

Matrix distribution will be made to following locations in quantities specified in EPDG #2, Attachment F-5.

EAL Classification Matrix Locations:

1. Control Room.
2. Simulator.
3. Emergency Operations Facility.
4. Technical Support Center.
5. Joint Information Center.
6. Emergency Preparedness Office.

ATTACHMENT 5 EAL GROUPS, CATEGORIES, AND SUBCATEGORIES (INFORMATION USE)

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
A - Abnormal Rad Release/Rad Effluent	1 - Off-Site Rad Conditions 2 - On-Site Rad Conditions and Spent Fuel Pool Events
H - Hazards And Other Conditions Affecting Plant Safety	1 - Natural or Destructive Phenomena 2 - Fire or Explosion 3 - Hazardous Gas 4 - Security 5 - Control Room Evacuation 6 - Judgment
E - ISFSI	None
MODES 1, 2, or 3:	
S - System Malfunction	1 - Loss of AC Power 2 - ATWS/Criticality 3 - Inability to Reach Shutdown Conditions 4 - Instrumentation 5 - Fuel Clad Degradation 6 - RCS Leakage 7 - Loss of DC Power 8 - Communications
F - Fission Product Barrier Degradation	None
MODES 4, 5, or DEF:	
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

CATEGORY A - ABNORMAL RAD RELEASE/RAD EFFLUENT

EAL Group: ANY (EALs in this category are applicable to any plant condition)

Many EALS are based on actual or potential degradation of fission product barriers because of the elevated potential for off-site radioactivity release. Degradation of fission product barriers though is not always apparent via non-radiological 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, off-site radiological conditions may result which require off-site 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. OFF-SITE RAD CONDITIONS

   Direct indication of effluent Radiation Monitoring Systems provides a rapid assessment mechanism to determine releases in excess of classifiable limits. Projected off-site doses, actual off-site field measurements or measured release rates via sampling indicate doses or dose rates above classifiable limits.

2. ON-SITE RAD CONDITIONS AND SPENT FUEL POOL EVENTS

   Sustained general area radiation levels in excess of those indicating loss of control of radioactive materials or those levels which may preclude access to vital plant areas also warrant emergency classification.

CATEGORY H - HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

EAL GROUP: ANY (EALS in this category are applicable to any plant condition)

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 earthquakes, tornados, high winds, and high/low river levels 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 vehicle crashes, missile impacts, internal flooding, 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 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 pre-determined 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.

CATEGORY E - ISFSI

EAL GROUP: ANY (the EAL in this category is applicable to any plant condition)

An Independent Spent Fuel Storage Installation (ISFSI) is a complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage. A significant amount of the radioactive material contained within a cask must escape its packaging and enter the biosphere for there to be a significant environmental effect resulting from an accident involving the dry storage of spent nuclear fuel. Formal off-site planning is not required because the postulated worst-case accident involving an ISFSI has insignificant consequences to the public health and safety.

An Unusual Event is declared 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.

A security event that leads to a potential loss of level of safety of the ISFSI is a classifiable event under Security Category EAL HU4.1.

Minor surface damage that does not affect storage cask boundary is excluded from the scope of these EALs.

CATEGORY S - SYSTEM MALFUNCTION

EAL GROUP: MODES 1, 2, OR 3

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 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 Technical Specifications if a limiting condition for operation (LCO) is not met.

4. INSTRUMENTATION

   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.

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 base-line levels (1% 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 Specification 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 Specification limits are utilized to indicate potential pipe cracks that may propagate to an extent threatening fuel clad, RCS, and containment integrity.

7. LOSS OF DC POWER

Loss of vital critical DC 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 VDC power sources.

8. COMMUNICATIONS

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

CATEGORY F - FISSION PRODUCT BARRIER DEGRADATION

EAL GROUP: MODE 1, 2, OR 3

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 zircaloy 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 3). "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.

The logic used for emergency classification based on fission product barrier monitoring should reflect 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 off-site 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 Specification 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 combination 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, an event for 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.

CATEGORY C - COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTION

EAL GROUP: MODES 4, 5, DEF

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 - Refueling, 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 on-site and off-site sources for 4160V emergency buses and loss of vital 125 VDC power sources.

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 vital critical DC 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 VDC power sources.

ATTACHMENT 6 EAL DEFINITIONS AND ACRONYMS (INFORMATION USE)

DEFINITIONS

Affecting Safe Shutdown

Event in-progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable hot or cold shutdown condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in hot shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is not "affecting safe shutdown".

Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in cold shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is "affecting safe shutdown".

Alert

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

Bomb

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

Civil Disturbance

A group of people violently protesting station operations or activities at the site.

Confinement Boundary

Is the barrier(s) between areas containing radioactive substances and the environment.

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 requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.

Emergency Action Level (EAL)

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

Emergency Classification Level (ECL)

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

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

Explosion

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

Extortion

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

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.

Flooding

Flooding, as used within the EALs, describes a condition where water is entering a room faster than installed equipment is capable of removal, resulting in a rise of water level within the room.

General Emergency

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

Hostage

Person(s) held as leverage against the station to ensure that demands will be met by the station.

Hostile Action

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

Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on CNS. Non-terrorism-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.

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 timeframes are specified, they shall apply.

Initiating Condition (IC)

One of a pre-determined subset of nuclear power plant conditions where either the potential exists for a radiological emergency or such an emergency has occurred.

Inoperable

Not able to perform its intended function.

Intruder

Person(s) present in a specified area without authorization.

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.

Independent Spent Fuel Storage Installation (ISFSI)

A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

Normal 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.

Operating Mode Applicability

The plant mode existing at the start of an event under which a particular EAL is applicable. These modes are defined in Technical Specifications Table 1.1-1. The Defueled or DEF mode referred to in some EALs is the condition where all fuel has been removed from the reactor vessel.

Note that the ISFSI EAL has no mode applicability.

Projectile

An object directed toward CNS that could cause concern for its continued operability, reliability, or personnel safety.

Protected Area

An area which normally encompasses all controlled areas within the security Protected Area fence as depicted in Technical Specifications, Figure 4.1-1, Site and Exclusion Area Boundaries and Low Population Zone.

Sabotage

Deliberate damage, mis-alignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. Equipment found tampered with or damaged due to malicious mischief may not meet the definition of sabotage until this determination is made by Security Supervision.

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.

Significant Transient

An unplanned event involving any of the following:

- Runback > 25% thermal power.
- Electrical load rejection > 25% full electrical load.
- Reactor scram.
- ECCS injection.
- Thermal power oscillations > 10%.

Site Area Emergency

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

Strike Action

Work stoppage within the Protected Area by a body of workers to enforce compliance with demands made on CNS. The strike action must threaten to interrupt Normal Plant Operations.

Sustained Wind

Sustained winds are of a prolonged duration and, therefore, do not include gusts. Sustained winds are not intermittent or of a transitory nature. Since the inauguration of the Automatic Surface Observation System (ASOS), the National Weather Service has adopted a 2 minute average standard for its sustained wind definition.

Unisolable

A breach or leak that cannot be promptly isolated.

Unplanned

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

Unusual Event

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

Valid

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

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. Examples of damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

Vital Area

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.

Acronyms

AC .... Alternating Current

ADS .... Automatic Depressurization System

APRM .... Average Power Range Meter

ATWS .... Anticipated Transient Without Scram

BIIT .... Boron Injection Initiation Temperature

BWR .... Boiling Water Reactor

CCW .... Component Cooling Water

CDE .... Committed Dose Equivalent

CFR .... Code of Federal Regulations

cps .... Counts per Second

CRD .... Control Rod Drive

CS .... Core Spray

CST .... Condensate Storage Tank

CTMT/CNMT .... Containment

DBA .... Design Basis Accident

DC .... Direct Current

Demin .... Demineralizer

DHRP .... Decay Heat Removal Pressure

DOT .... Department of Transportation

DW .... Drywell

DWSIL.................. Drywell Spray Initiation Limit

EAL .... Emergency Action Level

ECCS .... Emergency Core Cooling System

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

ERD .... Emergency RPV Depressurization

ESF .... Engineered Safety Feature

ESW .... Emergency Service Water

FAA .... Federal Aviation Administration

FBI .... Federal Bureau of Investigation

FEMA .... Federal Emergency Management Agency

FSAR................. Final Safety Analysis Report

ft .... Feet

gal .... Gallon(s)

GE .... General Emergency

GPM .... Gallons Per Minute

HCTL.................. Heat Capacity Temperature Limit

HCU .... Hydraulic Control Unit

HOO.................... Headquarters (NRC) Operations Officer

HPCI .... High Pressure Coolant Injection

H2 .... Hydrogen

hr .... Hour

HX .... Heat Exchanger

IC .... Initiating Condition

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 of Operation

LER..................... Licensee Event Report

LOCA .... Loss of Coolant Accident

LPCI.................. Low Pressure Coolant Injection

LWR.................. Light Water Reactor

MDRIR .... Minimum Debris Retention Injection Rate

MDSL................. Minimum Debris Submergence Level

MELLL .... Maximum Extended Load Line Limit

min.................... Minimum, minute

mR .... milliRoentgen

mRem .... milliRem

MSCP .... Minimum Steam Cooling Pressure

MSIV.................. Main Steam Isolation Valve

MSL .... Main Steam Line

MW .... Megawatt

N/A.................... Not applicable

NEI .... Nuclear Energy Institute

NESP.................. National Environmental Studies Project

NORAD.............. North American Aerospace Defense Command

NPP.................... Nuclear Power Plant

NR .... Narrow Range

NRC .... Nuclear Regulatory Commission

NSSS .... Nuclear Steam Supply System

NUMARC............ Nuclear Management and Resources Council

O_2........................ Oxygen

OBE .... Operating Basis Earthquake

OCA .... Owner Controlled Area

ODCM/ODAM....... Off-site Dose Calculation (Assessment) Manual

ORO.................. Off-site Response Organization

PA .... Protected Area

PAG .... Protective Action Guideline

PC .... Primary Containment

PCPL .... Primary Containment Pressure Limit

PMIS .... Plant Management Information System

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

R .... Roentgen

RB .... Reactor Building

RCC .... Reactor Control Console

RCIC .... Reactor Core Isolation Cooling

RCS .... Reactor Coolant System

rem .... Roentgen Equivalent Man

RETS.................. Radiological Effluent Technical Specifications

RHR .... Residual Heat Removal

RPS.................... Reactor Protection System

RPV.................... Reactor Pressure Vessel

RWCU .... Reactor Water Cleanup

SAG .... Severe Accident Guideline

SFP.................... Spent Fuel Pool

SGT .... Standby Gas Treatment

SBO .... Station Blackout

SLC.................... Standby Liquid Control

SPDS .... Safety Parameter Display System

SRO .... Senior Reactor Operator

SRV .... Safety Relief Valve

SSE .... Safe Shutdown Earthquake

TAF.................... Top of Active Fuel

TEDE.................. Total Effective Dose Equivalent

TSC .... Technical Support Center

UE .... Notification Of Unusual Event

USAR .... Updated Final Safety Analysis Report

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

ATTACHMENT 7 INFORMATION SHEET (INFORMATION USE)

1. PURPOSE

1.1 Procedure contains the instructions necessary for classification of emergencies consistent with the NRC approved EAL classification scheme. Also, included are the explanations and rational for the scheme, the detailed bases for the EALs, and controls required for the EAL Classification Matrix which is the primary tool used to determine when classification criteria are exceeded.

1.2 Procedure provides the formal set of threshold conditions necessary to classify an event at CNS into one of the four emergency classifications described in NUREG-0654, NEI 99-01, Revision 5, and the CNS Emergency Plan.©[2]

2. PRECAUTIONS AND LIMITATIONS

2.1 Assessment, classification, and declaration of an emergency condition shall be completed within 15 minutes after initial availability of indications to plant Operators that an EAL has been exceeded provided that:

2.1.1 Implementation of response actions required to protect public health and safety are not delayed; and,

2.1.2 Any delay in declaration does not deny State and Local authorities opportunity to implement measures necessary to protect public health and safety.

2.1.3 Classifying and declaration of an emergency is a non-delegable responsibility of Emergency Director. Although additional input in these decisions is encouraged, completion of timely and accurate performance of these activities rests solely with Emergency Director.

3. REFERENCES

3.1 CODES AND STANDARDS

3.1.1 10CFR50.48, Fire Protection.

3.1.2 10CFR50.72, Immediate Notification Requirements for Operating Nuclear Power Reactors.

3.1.3 10CFR72.32, Emergency Plan.

3.1.4 NEI 99-01, Revision 5, Methodology for the Development of Emergency Action Levels, February 2008 (ADAMS Accession Number ML080450149).

3.1.5 NPPD Emergency Plan for CNS.

3.1.6 NUREG-0654, Revision 1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants.

3.1.7 NUREG-1022, Event Reporting Guidelines 10CFR50.72 and 50.73, Revision 2.

3.2 PROCEDURES

3.2.1 Instrument Operating Procedure 4.12, Seismic Instrumentation.

3.2.2 Emergency Plan Implementing Procedure 5.7.2, Emergency Director EPIP.

3.2.3 Emergency Plan Implementing Procedure 5.7.16, Release Rate Determination.

3.2.4 Emergency Plan Implementing Procedure 5.7.17, CNS-DOSE Assessment.

3.2.5 Emergency Plan Implementing Procedure 5.7.17.1, Dose Assessment (Manual).

3.3 MISCELLANEOUS

3.3.1 ADAMS Accession No. ML100080231, Cooper Nuclear Station - Change to Emergency Action Level Scheme (TAC NO. ME0849). Document ID NRC2010008.

3.3.2 ADAMS Accession No. ML14055A023, Cooper Nuclear Station - Issuance of Amendment Regarding Transition to Risk-Informed, Performance-Based Fire Protection Program in Accordance With 10CFR50.48(c) (TAC No.ME8551).

3.3.3 RIS 2007-02, Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events.

3.3.4 ©[1] NRC Bulletin 2005-02 (Commitment Number NLS2005080-02), EALs reflect information provided in Attachment 2 of bulletin. Commitment affects EALs HA4.1, HS4.1, and HG4.1.

3.3.5 ©[2] IR 81-013 (Commitment Number 811217-01-07), Develop Functional Procedure for Accident Classification. Commitment affects Attachment 7, Step 1.2.

3.3.6 ©[3] IR 92-014 (Commitment Number 921110-02-03), Revise Guidance Documents for "Low Threshold" Core Damage Events. Commitment affects Attachment 3, Threshold 4, 5, and 6 CNS Basis.

ATTACHMENT 8 MATRIX BASIS CROSS-REFERENCE (INFORMATION USE)

CAT	SUB	GE		SAE		ALERT		UE	
A	1	AG1.1	40	AS1.1	34	AA1.1	26	AU1.1	18
		AG1.2	42	AS1.2	36	AA1.2	29	AU1.2	21
		AG1.3	44	AS1.3	38	AA1.3	32	AU1.3	24
	2					AA2.1	51	AU2.1	46
						AA2.2	53	AU2.2	49
	3					AA3.1	55		
H	1					HA1.1	130	HU1.1	118
						HA1.2	134	HU1.2	122
						HA1.3	137	HU1.3	124
						HA1.4	140	HU1.4	126
						HA1.5	143	HU1.5	128
						HA1.6	145		
	2					HA2.1	152	HU2.1	148
								HU2.2	150
	3					HA3.1	161	HU3.1	154
								HU3.2	159
	4	HG4.1	174	HS4.1	172	HA4.1	169	HU4.1	166
	5			HS5.1	177	HA5.1	176		
	6	HG6.1	185	HS6.1	183	HA6.1	181	HU6.1	179
E								EU1.1	243
S	1	SG1.1	202	SS1.1	197	SA1.1	192	SU1.1	187
	2	SG2.1	216	SS2.1	213	SA2.1	209	SU2.1	208
	3							SU3.1	220
	4			SS4.1	227	SA4.1	224	SU4.1	221
	5							SU5.1	231
								SU5.2	233
	6							SU6.1	234
	7			SS7.1	236				

CAT	SUB	GE		SAE		ALERT		UE	
	8							SU8.1	240
F	1	FG1.1	248	FS1.1	246	FA1.1	245	FU1.1	244
C	1					CA1.1	62	CU1.1	57
	2	CG2.1	85	CS2.1	77	CA2.1	74	CU2.1	67
		CG2.2	91	CS2.2	79			CU2.2	69
				CS2.3	81			CU2.3	71
	3					CA3.1	105	CU3.1	99
								CU3.2	102
	4							CU4.1	109
	5							CU5.1	112
	6							CU6.1	114

Fission Product Barrier Matrix												
	Fuel Clad Barrier				Reactor Coolant Sys Barrier				Primary Cont. Barrier			
	Loss		Pot. Loss		Loss		Pot. Loss		Loss		Pot. Loss	
A	1	252	8	254	10	264					25	280
B					11	267			19	282	26	284
									20	283	27	285
											28	287
C					12	269	16	271	21	288		
							17	272	22	290		
									23	291		
D					13	276						
E	2	257										
	3	258			14	277					29	294
	4	259										
	5	260										
	6	261										
F	7	262	9	263	15	278	18	279	24	295	30	297

Attachment 1

Report of Change and Summary of 50.54(q) Analyses – Emergency Plan Implementing Procedure 5.7.1, Revision 54

Change 1 Description

Revision 54 of Emergency Implementing Procedure (EPIP) 5.7.1 revised the technical basis for General Emergency, Emergency Action Level (EAL) SG2.1 and Fission Product Barrier Matrix Thresholds 8 and 10 for Fuel Clad Barrier Potential Loss and Reactor Coolant System Barrier Loss. The change removed the 15 minute time clock and associated reset criteria for restoring and maintaining reactor pressure vessel (RPV) level and replaced it with the statement "Determination of inability to restore and maintain RPV level is based on actions driven by EOPs to restore level."

Change 1 Summary of Analysis (10 CFR 50.54(q) evaluation)

Licensing Basis Affected by Change:

EPIP 5.7.1 contains the Nuclear Regulatory Commission approved EAL classification scheme which is based on NEI 99-01, Revision 5, Methodology for Development of Emergency Action Levels. The Cooper Nuclear Station (CNS) Emergency Plan (E-Plan) lists the EALs but does not contain the EAL technical basis. The following E-Plan sections were relevant to the change:

E-Plan, Section 4, discusses that CNS maintains the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an EAL has been exceeded and that CNS shall promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level.

EAL SG2.1 is listed in E-Plan Table 4.1-4.

The change does not affect the above E-Plan requirements.

How Change Complies with Regulations and Previous Commitments:

10 CFR 50.47(b)(4) requires that a standard emergency classification and action level scheme is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.

10 CFR 50, Appendix E, Section IV.C.2, requires that nuclear power reactor licensees establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and shall promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level.

The change continues to comply with the above regulatory requirements and does not affect regulatory commitments.

Affected Emergency Planning Functions/Impact on Effectiveness of Emergency Planning Functions:

10 CFR 50.47(b)(4); Function - A standard scheme of emergency classification and action levels is in use.

This change aligns with NEI 99-01, Revision 5, as the technical basis for NEI 99-01 does not include a time clock philosophy associated with EALs, or Fission Product Barrier criteria that rely on the criteria "cannot be restored or maintained." The change complies with regulatory requirements and is not considered a reduction in the effectiveness of the CNS E-Plan.

Change 2 Description

The manual methodology sections for performing dose assessment have been removed from EPIP 5.7.17, CNS-DOSE Assessment, and placed into new EPIP, 5.7.17.1, Dose Assessment (Manual). This new EPIP will be used strictly for manual performance of dose assessment.

Core damage estimate information was moved from EPIP 5.7.17 to new EPIP 5.7.17.1. As such, a conforming change was made to EPIP 5.7.1 to reference new EPIP 5.7.17.1 for core damage estimation.

Change 2 Summary of Analysis (10 CFR 50.54(q) evaluation)

The 10 CFR 50.54(q) evaluation for new EPIP 5.7.17.1 is summarized below. This evaluation encompasses the change made to EPIP 5.7.1.

Licensing Basis Affected by Change:

CNS E-Plan, Section 1.3, provides the definition of Class 'A' Dose Assessment Model.

E-Plan, Section 1.17, provides the definition of the Plume Exposure Pathway.

E-Plan, Section 1.20, provides the definition of Protective Action Guides.

E-Plan, Section 4.2.4, discusses that the dose rate and integrated dose are based on duration of release, release rates, meteorological data, and atmospheric dispersion factors.

E-Plan, Table 6.3-1, discusses the means to obtain data for dose assessment (computer model, meteorological output, effluent monitors, etc.) and also states that manual dose assessment techniques are available in the event computer programs are unavailable.

E-Plan, Section 6.3.3, describes the two techniques for performing dose projections, CNS-DOSE (computerized class 'A' model) and hand calculation (manual method).

E-Plan, Section 7.2.3, discusses that the Emergency Offsite Facility and Technical Support Center have the capability to assess meteorological data, current plant conditions, and release rate data to determine projected downwind doses.

E-Plan, Appendix A, lists procedures that implement the E-Plan and provides a description, including EPIPs 5.7.17, CNS-DOSE Assessment, and 5.7.20, Protective Action Recommendations.

The E-Plan definitions and methods described above are not affected and continue to be met.

How Change Complies with Regulations and Previous Commitments:

10 CFR 50.47(b)(9), discusses that adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

10 CFR 50.47(b)(15), discusses that radiological emergency response training is provided to those who may be called on to assist in an emergency.

10 CFR 50, Appendix E, Section IV.B.1, discusses that the means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described.

The change continues to comply with the both regulatory requirements and commitments. Commitment references were moved to new EPIP 5.7.17.1 as appropriate.

Affected Emergency Planning Functions/Impact on Effectiveness of Emergency Planning Functions:

10 CFR 50.47(b)(9); Function - Methods, systems, and equipment for radiological releases are in use.

10 CFR 50.47(b)(15); Training is provided to emergency responders.

The change does not impact the ability to perform dose assessment and develop protective action recommendations (PAR) relative to the dose. The methods for assessing dose and training provided to emergency response organization personnel that perform dose assessment or develop, review, and approve PARs is not impacted.

Other Changes

Other changes were also made to EPIP 5.7.1 that were determined to be editorial or did not affect an emergency planning element/function. These changes did not require a 10 CFR 50.54(q) evaluation.

Attachment 2

**Report of Change and Summary of 50.54(q) Analysis –
Emergency Plan Implementing Procedure 5.7.8, Revision 27**

Change Description

Revision 27 to Emergency Implementing Procedure (EPIP) 5.7.8 moved guidance from a position instruction manual for Operational Support Center (OSC) Supervisor into Attachment 1, OSC Activation and Operation, and Attachment 2, OSC Briefing Guide, in order to provide a single source of instruction.

Change Summary of Analysis (10 CFR 50.54(q) evaluation)

Licensing Basis Affected by Change:

Cooper Nuclear Station (CNS) Emergency Plan (E-Plan), Section 1.16, provides the definition of the OSC.

E-Plan, Section 5.2.4, discusses the responsibilities of the OSC Supervisor.

E-Plan, Section 7.2.2, describes the OSC.

E-Plan, Appendix A, lists procedures implementing the E-Plan, including EPIP 5.7.8.

The change continues to comply with assigned responsibilities of the Emergency Response Organization (ERO) and continues to ensure that the capability to respond and to augment staff on a continuing basis in accordance with the E-Plan is maintained.

How Change Complies with Regulations and Previous Commitments:

10 CFR 50, Appendix E, Section IV.1, requires that emergency plans shall contain but not be limited to, information needed to demonstrate compliance with the elements of organization for coping with radiological emergencies, assessment actions, activation of emergency organization, notification procedures, emergency facilities and equipment, training, maintaining emergency preparedness, recovery, and onsite protective actions during hostile action.

NUREG-0654, FEMA-REP-1, Section II, Planning Standard A, discusses primary responsibilities for emergency response by the licensee, and by State and local organizations within the emergency planning zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.

This change continues to comply with regulatory requirements and does not conflict with regulatory commitments.

Affected Emergency Planning Functions/Impact on Effectiveness of Emergency Planning Functions:

10 CFR 50.47(b)(1); Function - Responsibility for emergency response is assigned.

10 CFR 50.47(b)(1); Function - The response organization has the staff to respond and to augment staff on a continuing basis in accordance with the emergency plan.

The change will improve ERO performance by providing a single source to coordinate the activation and operation of the OSC which will enable the OSC Supervisor to more efficiently execute E-Plan responsibilities. This change continues to meet the planning standard/functions described above, and does not constitute a reduction in effectiveness of the CNS E-Plan.

Other Changes

Other changes were also made to EPIP 5.7.8 that were determined to be editorial. These changes did not require a 10 CFR 50.54(q) evaluation.

Attachment 3

**Report of Change and Summary of 50.54(q) Analyses –
Emergency Plan Implementing Procedure 5.7.16, Revision 25**

Change 1 Description

Emergency Plan Implementing Procedure (EPIP) 5.7.16, Revision 25, is considered a major rewrite to meet new format and performance guidance provided in the station's procedure writer's guide. Guidance pertaining to dose assessment was also clarified and/or reformatted throughout the EPIP.

Change 1 Summary of Analysis (10 CFR 50.54(q) evaluation)

Licensing Basis Affected by Change:

Cooper Nuclear Station (CNS) Emergency Plan (E-Plan), Section 4.2, discusses the methods for off-site radiological assessment.

E-Plan, Section 7.5, describes the equipment available for evaluation and assessment of emergency conditions. Specifically, Section 7.5.4.1 discusses the Effluent Release Point Monitors, and Section 7.5.5.1 discusses the Steam Jet Air Ejector Monitors.

The change continues to meet the requirements as described in the CNS E-Plan.

How Change Complies with Regulations and Previous Commitments:

10 CFR 50, Appendix E, Section IV.B.1, discusses that means used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described.

10 CFR 50, Appendix E, Section IV.E.2, discusses that adequate provisions shall be made and described for emergency facilities and equipment including equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment.

This change continues to comply with the above emergency preparedness regulations and previous regulatory commitments.

Affected Emergency Planning Functions/Impact on Effectiveness of Emergency Planning Functions:

10 CFR 50.47(b)(9); Function - Methods, systems, and equipment for assessment of radioactive releases are in use.

The revised guidance will continue to meet the requirements of the E-Plan and does not add new responsibilities for emergency response or add new training for emergency responders. The change does not affect the above planning function and is not considered a reduction in effectiveness of the CNS E-Plan.

Change 2 Description

As part of the complete rewrite made to EPIP 5.7.16, a requirement to ensure the availability of various radiation monitors, a scientific calculator, and Steam Jet Air Ejector Monitors was deleted. This step was considered unnecessary as the majority of the equipment is part of the plant's design.

Change 2 Summary of Analysis (10 CFR 50.54(q) evaluation)

Licensing Basis Affected by Change:

CNS E-Plan, Section 4.2, discusses the methods for off-site radiological assessment.

E-Plan, Section 7.5, describes the equipment available for evaluation and assessment of emergency conditions. Specifically, Section 7.5.4.1 discusses the Effluent Release Point Monitors, and Section 7.5.5.1 discusses the Steam Jet Air Ejector Monitors.

The change continues to meet the requirements as described in the CNS E-Plan.

How Change Complies with Regulations and Previous Commitments:

10 CFR 50, Appendix E, Section IV.B.1, discusses that means used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described.

10 CFR 50, Appendix E, Section IV.E.2, discusses that adequate provisions shall be made and described for emergency facilities and equipment including equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment.

This change continues to comply with the above emergency preparedness regulations and previous regulatory commitments.

Affected Emergency Planning Functions/Impact on Effectiveness of Emergency Planning Functions:

10 CFR 50.47(b)(9); Function - Methods, systems, and equipment for assessment of radioactive releases are in use.

The revised guidance will continue to meet the requirements of the E-Plan and does not add new responsibilities for emergency response or add new training for emergency responders.

The change does not affect the above planning function and is not considered a reduction in effectiveness of the CNS E-Plan.

Other Changes

Other changes were also made to the EPIP that were determined to be editorial. These changes did not require a 10 CFR 50.54(q) evaluation.

Attachment 4

**Report of Change and Summary of 50.54(q) Analysis –
Emergency Plan Implementing Procedure 5.7.17.1, Revision 0**

Change 1 Description

The manual methodology sections for performing dose assessment have been removed from Emergency Plan Implementing Procedure (EPIP) 5.7.17, CNS-DOSE Assessment, and placed into new EPIP, 5.7.17.1, Dose Assessment (Manual). This new EPIP will be used for the manual performance of dose assessment. Step sequence was restructured based on priority and frequency of the manual methods and also established a method to correct a manual calculation to a field team sample where appropriate.

Change 1 Summary of Analysis (10 CFR 50.54(q) evaluation)

Licensing Basis Affected by Change:

Cooper Nuclear Station (CNS) Emergency Plan (E-Plan), Section 1.3, provides the definition of Class 'A' Dose Assessment Model.

E-Plan, Section 1.17, provides the definition of the Plume Exposure Pathway.

E-Plan, Section 1.20, provides the definition of Protective Action Guides.

E-Plan, Section 4.2.4, discusses that the dose rate and integrated dose are based on duration of release, release rates, meteorological data, and atmospheric dispersion factors.

E-Plan, Table 6.3-1, discusses the means to obtain data for dose assessment (computer model, meteorological output, effluent monitors, etc.) and also states that manual dose assessment techniques are available in the event computer programs are unavailable.

E-Plan, Section 6.3.3, describes the two techniques for performing dose projections, CNS-DOSE (computerized class 'A' model) and hand calculation (manual method).

E-Plan, Section 7.2.3, discusses that the Emergency Offsite Facility and Technical Support Center have the capability to assess meteorological data, current plant conditions, and release rate data to determine projected downwind doses.

E-Plan, Appendix A, lists procedures that implement the E-Plan and provides a description of each procedure, including EPIPs 5.7.17, CNS-DOSE Assessment, and 5.7.20, Protective Action Recommendations.

The E-Plan definitions and methods described above are not affected and continue to be met.

How Change Complies with Regulations and Previous Commitments:

10 CFR 50.47(b)(9) discusses that adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

10 CFR 50.47(b)(15) discusses that radiological emergency response training is provided to those who may be called on to assist in an emergency.

10 CFR 50, Appendix E, Section IV.B.1, discusses that the means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described.

The change continues to comply with both regulatory requirements and commitments. The commitment references associated with performing dose assessment manually were relocated from EPIP 5.7.17 to this new EPIP.

Affected Emergency Planning Functions/Impact on Effectiveness of Emergency Planning Functions:

10 CFR 50.47(b)(9); Function - Methods, systems, and equipment for radiological releases are in use.

10 CFR 50.47(b)(15); Training is provided to emergency responders.

The change does not impact the ability to perform dose assessment and develop protective action recommendations (PAR) relative to the dose. The methods for assessing dose and training provided to emergency response organization personnel that perform dose assessment or develop, review, and approve PARs is not impacted. There is no reduction in the effectiveness of the CNS E-Plan as a result of this change.

Other Changes

Other changes were also made to steps and sections moved from EPIP 5.7.17 to clarify guidance and improve usability. These changes were determined to be editorial and did not require a 10 CFR 50.54(q) evaluation.

Attachment 5

Report of Change and Summary of 50.54(q) Analyses – Emergency Plan Implementing Procedure 5.7.17, Revision 45

Change 1 Description

Emergency Plan Implementing Procedure (EPIP) 5.7.17, Revision 45, is considered a major rewrite to meet new format and performance guidance provided in the station's procedure writer's guide. Flowcharts were also added to provide rapidly progressing severe accident guidance, sequence changes made to better align to the use of the Plant Management Information System, instruction added for using a personal computer instead of an information display terminal, and reference to new EPIP 5.7.17.1 added for dose assessment manual calculations. Additionally, multiple tables are were added to the EPIP for determining if the release is through the Reactor Building and filtered through the Standby Gas Treatment system, and also for adjustment of the release to actual downwind values.

Change 1 Summary of Analysis (10 CFR 50.54(q) evaluation)

Licensing Basis Affected by Change:

Cooper Nuclear Station (CNS) Emergency Plan (E-Plan), Section 1.3, provides the definition of Class 'A' Dose Assessment Model.

E-Plan, Section 1.17, provides the definition of the Plume Exposure Pathway.

E-Plan, Section 1.20, provides the definition of Protective Action Guides.

E-Plan, Section 4.2.4, discusses that the dose rate and integrated dose are based on duration of release, release rates, meteorological data, and atmospheric dispersion factors.

E-Plan, Table 6.3-1, discusses the means to obtain data for dose assessment (computer model, meteorological output, effluent monitors, etc.) and also states that manual dose assessment techniques are available in the event computer programs are unavailable.

E-Plan, Section 6.3.3, describes the two techniques for performing dose projections, CNS-DOSE (computerized class 'A' model) and hand calculation (manual method).

E-Plan, Section 7.2.3, discusses that the Emergency Offsite Facility and Technical Support Center have the capability to assess meteorological data, current plant conditions, and release rate data to determine projected downwind doses.

E-Plan, Appendix A, lists procedures that implement the E-Plan and provides a description of each procedure, including EPIPs 5.7.17, CNS-DOSE Assessment, and 5.7.20, Protective Action Recommendations.

The E-Plan definitions and methods described above are not affected and continue to be met.

How Change Complies with Regulations and Previous Commitments:

10 CFR 50.47(b)(9) discusses that adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

10 CFR 50.47(b)(15) discusses that radiological emergency response training is provided to those who may be called on to assist in an emergency.

10 CFR 50, Appendix E, Section IV.B.1, discusses that the means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described.

The change continues to comply with both regulatory requirements and commitments. The commitment reference associated with performing dose assessment manually was relocated to new EPIP 5.7.17.1.

Affected Emergency Planning Functions/Impact on Effectiveness of Emergency Planning Functions:

10 CFR 50.47(b)(9); Function - Methods, systems, and equipment for radiological releases are in use.

10 CFR 50.47(b)(15); Training is provided to emergency responders.

The change does not impact the ability to perform dose assessment and develop protective action recommendations (PAR) relative to the dose. The methods for assessing dose and training provided to emergency response organization personnel that perform dose assessment or develop, review, and approve PARs is not impacted. There is no reduction in the effectiveness of the CNS E-Plan as a result of this change.

Change 2 Description

As part of the complete rewrite made to EPIP 5.7.17, instruction was deleted associated with using specific keyboard buttons when using the dose assessment program. This instruction is not needed for proper operation of the program.

Change 2 Summary of Analysis (10 CFR 50.54(q) evaluation)

Licensing Basis Affected by Change:

CNS E-Plan, Section 1.3, provides the definition of Class 'A' Dose Assessment Model.

E-Plan, Section 1.17, provides the definition of the Plume Exposure Pathway.

E-Plan, Section 1.20, provides the definition of Protective Action Guides.

E-Plan, Section 4.2.4, discusses that the dose rate and integrated dose are based on duration of release, release rates, meteorological data, and atmospheric dispersion factors.

E-Plan, Table 6.3-1, discusses the means to obtain data for dose assessment (computer model, meteorological output, effluent monitors, etc.) and also states that manual dose assessment techniques are available in the event computer programs are unavailable.

E-Plan, Section 6.3.3, describes the two techniques for performing dose projections, CNS-DOSE (computerized class 'A' model) and hand calculation (manual method).

E-Plan, Section 7.2.3, discusses that the Emergency Offsite Facility and Technical Support Center have the capability to assess meteorological data, current plant conditions, and release rate data to determine projected downwind doses.

E-Plan, Appendix A, lists procedures that implement the E-Plan and provides a description of each procedure, including EPIP 5.7.17, CNS-DOSE Assessment, and 5.7.20, Protective Action Recommendations.

The E-Plan definitions and methods described above are not affected and continue to be met.

How Change Complies with Regulations and Previous Commitments:

10 CFR 50.47(b)(9), discusses that adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

10 CFR 50.47(b)(15), discusses that radiological emergency response training is provided to those who may be called on to assist in an emergency.

10 CFR 50, Appendix E, Section IV.B.1, discusses that the means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described.

The change continues to comply with regulatory requirements as the assessment and training methods have not been modified. One commitment reference associated with performing dose assessment manually was relocated to new EPIP 5.7.17.1.

Affected Emergency Planning Functions/Impact on Effectiveness of Emergency Planning Functions:

10 CFR 50.47(b)(9); Function - Methods, systems, and equipment for radiological releases are in use.

10 CFR 50.47(b)(15); Training is provided to emergency responders.

The change does not impact the ability to perform dose assessment and develop protective action recommendations relative to the dose. Certain keyboard instructions were needed at the time of initial implementation; however the instruction is no longer required and will reduce the time spent reading and performing a step that is common knowledge with the users. There is no reduction in the effectiveness of the CNS E-Plan as a result of this change.

Other Changes

Other changes were also made to the EPIP that were determined to be editorial. These changes did not require a 10 CFR 50.54(q) evaluation.

Attachment 6

**Report of Change and Summary of 50.54(q) Analysis –
Emergency Plan Implementing Procedure 5.7.20, Revision 27**

Change Description

Emergency Plan Implementing Procedure (EPIP) 5.7.20, Revision 27, is considered a major rewrite to meet new format and performance guidance provided in the station's procedure writer's guide. Steps and sections were relocated throughout the procedure and instruction added to provide clarification. This change enhances the EPIP and will reduce the potential for human performance errors.

Change Summary of Analysis (10 CFR 50.54(q) evaluation)

Licensing Basis Affected by Change:

Cooper Nuclear Station (CNS) Emergency Plan (E-Plan), Sections 1.19 and 1.20, provide the definitions for protective actions and protective action guides, respectively.

E-Plan, Section 4.1.4 and Table 4.1-8 (Item 3), provides an overview of a General Emergency including the protective action recommendations (PAR) taken when one is declared.

E-Plan, Section 6.5, discusses the roles of the Radiological Control Manager and Emergency Director in determining PARs.

E-Plan, Appendix A, lists procedures that implement the E-Plan and provides a description of each procedure, including EPIPs 5.7.17, CNS-DOSE Assessment, and 5.7.20, Protective Action Recommendations.

The change does not impact the above E-Plan requirements.

How Change Complies with Regulations and Previous Commitments:

10 CFR 50, Appendix E, Section IV.B.1, requires that the means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within and outside the site boundary to protect health and safety.

The change does not conflict with regulatory commitments and continues to comply with the above regulation.

Affected Emergency Planning Functions/Impact on Effectiveness of Emergency Planning Functions:

10 CFR 50.47(b)(10); Function - A range of PARs is available for implementation during emergencies.

The change continues to provide capabilities and resources necessary to prepare for, and respond to a radiological emergency as required by the CNS E-Plan and emergency preparedness regulations. The change does not alter EPIP technical content, add new responsibilities for emergency response, and does not add new training required for emergency responders. The change does not represent a reduction in effectiveness of the CNS E-Plan.

Other Changes

Other changes were also made to the EPIP that were determined to be editorial. These changes did not require a 10 CFR 50.54(q) evaluation.

Enclosure 1

Emergency Plan Implementing Procedure 5.7.1, Revision 54

COOPER NUCLEAR STATION

Operations Manual

Emergency Preparedness

EMERGENCY PLAN IMPLEMENTING PROCEDURE

5.7.1

EMERGENCY CLASSIFICATION

Level of Use:	**MULTIPLE**
Quality:	**QAPD RELATED**
Effective Date:	**3/28/16**
Approval Authority:	**ITR-RDM**
Procedure Owner:	**EMERG PREP DRILL SCENARIO COORD**

TABLE OF CONTENTS

1. ENTRY CONDITIONS (REFERENCE USE)

   1.1 An Emergency Operation Procedure has been initiated; or

   1.2 An unusual occurrence has taken place at or near site.

2. INITIAL CLASSIFICATION AND DECLARATION (REFERENCE USE)

   2.1 AFTER recognition of off-normal event,
   THEN SM shall:

   2.1.1 **COMPARE** event to EALs on EAL Classification Matrix.

   2.1.2 IF more than one EAL of different classification levels is reached (i.e., EAL for ALERT and EAL for SITE AREA EMERGENCY),
   THEN **SELECT** EAL for most severe emergency classification.

   2.1.3 IF event appears to meet an EAL and time permits,
   THEN **REFER** to Attachment 2 or 3 for further explanation and guidance.

   2.1.4 IF determined EAL is met,
   THEN **PERFORM** following within 15 minutes:

   2.1.4.1 **ASSUME** Emergency Director Responsibilities.

   2.1.4.2 **DECLARE** emergency.

   2.1.4.3 Enter EPIP 5.7.2.

   2.1.4.4 As time permits, **RECORD** emergency class, time of declaration, and EAL number in CNS Operations Log.

3. RECLASSIFICATION (REFERENCE USE)

NOTE – Specific direction for upgrading or terminating an event is given via EPIP 5.7.2.

3.1 AFTER SM has been relieved of Emergency Director duties, THEN SM shall bring changing emergency conditions to attention of Emergency Director.

3.2 **COMPARE** changing emergency conditions to EALs on EAL Classification Matrix.

3.3 IF more than one EAL of different classification levels is reached (i.e., EAL for ALERT and EAL for SITE AREA EMERGENCY), THEN **SELECT** EAL for most severe emergency classification.

3.4 IF emergency condition appears to meet an EAL and time permits, THEN **REFER** to Attachment 2 or 3 for further explanation and guidance as necessary.

3.5 IF GENERAL EMERGENCY has been declared, THEN consultation with state authorities and NRC should occur prior to reclassification or termination of event.

3.6 IF determined new EAL is met, THEN:

3.6.1 **DECLARE** emergency.

3.6.2 **ENTER** appropriate section of EPIP 5.7.2.

3.6.3 **RECORD** emergency class, time of declaration, and EAL number in applicable Emergency Response Facility (ERF) Log.

4. MISSED EAL CLASSIFICATION (REFERENCE USE)

4.1 IF determined condition existed which met an EAL but no emergency was declared and basis for Emergency Class no longer exists (e.g., condition occurred yesterday but was not caught at time or condition cleared before classification could be made and emergency response is no longer needed),
THEN:

4.1.1 **NOTIFY** NRC within 1 hour of discovery of undeclared (or misclassified) event per Procedure 2.0.5.

4.1.2 **INFORM** Emergency Preparedness Manager of details.

4.1.2.1 Emergency Preparedness Staff member **CONTACT** Responsible State and Local Governmental Agencies.

a. **PROVIDE** details of undeclared event.

ATTACHMENT 1 EAL SCHEME EXPLANATION AND RATIONALE (INFORMATION USE)

1. PURPOSE

1.1 This attachment along with Attachments 2, 3, 5, 6, and 7 provide an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Scheme for Cooper Nuclear Station (CNS). It should be used to facilitate review of the CNS EALs, provide historical documentation for future reference, and serve as a resource for training. Decision-makers responsible for implementation of the EAL Scheme may use these attachments as a technical reference in support of EAL interpretation.

1.2 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.

1.3 This procedure, its attachments, and the EAL Classification Matrix are controlled pursuant to 10CFR50.54(q).

2. DISCUSSION

2.1 BACKGROUND

2.1.1 An EAL is a pre-determined, site specific, observable threshold for a plant Initiating Condition (IC) that places the plant in a given Emergency Classification Level (ECL). An EAL can be: an instrument reading; an equipment status indicator; a measurable parameter (on-site or off-site); a discrete, observable event; results of analyses; entry into specific emergency operating procedures; or another phenomenon which, if it occurs, indicates entry into a particular emergency classification level. EALs are utilized to classify emergency conditions defined in the CNS Emergency Plan.

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

2.1.3 NEI 99-01 (NUMARC/NESP-007), Revision 5, was the most recently accepted methodology for development of Emergency Action Levels approved by the NRC when CNS upgraded this classification scheme.

2.1.4 Using NEI 99-01, Revision 5, CNS conducted an EAL implementation upgrade project that produced the EALs discussed in this procedure. CNS has matched plant features and safety system designs that are unique to CNS to the generic NEI 99-01, Revision 5, guidance. Should CNS implement any design differences not addressed in NEI 99-01, Revision 5, CNS will have to consider the applicable failure mechanisms involved when assessing the impact on the site specific EALs.

2.2 FISSION PRODUCT BARRIERS

2.2.1 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.

2.2.2 The primary fission product barriers are:

2.2.2.1 Fuel Clad (FC): The Fuel Clad barrier consists of the zircaloy fuel bundle tubes that contain the fuel pellets.

2.2.2.2 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.

2.2.2.3 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.

2.3 EMERGENCY CLASSIFICATION BASED ON FISSION PRODUCT BARRIER DEGRADATION

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

2.3.1.1 Unusual Event: Any loss or any potential loss of Primary Containment.

2.3.1.2 Alert: Any loss or any potential loss of either Fuel Clad or RCS.

2.3.1.3 Site Area Emergency: Loss or potential loss of any two barriers.

2.3.1.4 General Emergency: Loss of any two barriers and loss or potential loss of third barrier.

2.4 EAL RELATIONSHIP TO EOPS

2.4.1 Where possible, the EALs have been made consistent with and utilize the conditions defined in the CNS 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.

2.5 SYMPTOM-BASED VS. EVENT-BASED APPROACH

2.5.1 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 pre-determined symptoms can be ascribed have also been utilized as EALs since they may be indicative of potentially more serious conditions not yet fully realized.

2.6 EAL ORGANIZATION

2.6.1 The CNS EAL scheme includes the following features:

2.6.1.1 Division of the EAL set into three broad groups.

a. 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 Modes 1, 2, or 3 - this group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Startup, or Power Operation mode.

3. EALs applicable only under Modes 4, 5, or Defueled - this group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling, or Defueled Mode.

b. The purpose of the groups is to avoid review of EALs that cannot be applicable in the current operating mode of the plant. 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.

2.6.1.2 Within each of the above three groups, assignment of EALs to categories/subcategories.

a. 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 CNS EAL categories/subcategories are contained in Attachment 5.

2.6.1.3 The primary tool for determining the emergency classification level is the EAL classification matrix. The user of the EAL classification matrix 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 2.7 and 2.8 of this attachment, and Attachments 2 and 3 for such information.

2.7 TECHNICAL BASES INFORMATION

2.7.1 EAL Technical Bases are provided in Attachment 2 for each EAL according to EAL group, EAL category (A, C, H, S, E, and F), 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:

2.7.1.1 CATEGORY LETTER AND TITLE.

2.7.1.2 SUBCATEGORY NUMBER AND TITLE.

2.7.1.3 INITIATING CONDITION (IC).

2.7.1.4 SITE-SPECIFIC DESCRIPTION OF THE GENERIC IC GIVEN IN NEI 99-01.

2.7.1.5 EAL IDENTIFIER (ENCLOSED IN RECTANGLE)

a. Each EAL is assigned a unique identifier to support accurate communication of the emergency classification to on-site and off-site personnel. Four characters define each EAL identifier:

   1. First character (letter): Corresponds to the EAL category as described in Attachment 5 (A, C, H, S, E, or F).

   2. Second character (letter): The emergency classification (G, S, A, or U).

   3. Third character (number): Subcategory number within the given category.

      a) 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.

      a) If the subcategory has only one EAL, it is given the number one (1).

2.7.1.6 CLASSIFICATION (ENCLOSED IN RECTANGLE)

a. Unusual Event (U), Alert (A), Site Area Emergency (S), or General Emergency (G).

2.7.1.7 EAL (ENCLOSED IN RECTANGLE)

a. Exact wording of the EAL as it appears in the EAL classification matrix.

2.7.1.8 MODE APPLICABILITY

a. One or more of the following plant operating conditions comprise the Mode to which each EAL is applicable: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refueling, D - Defueled (DEF), All or N/A - Not Applicable (see Section 2.8 for operating Mode definitions.)

2.7.1.9 NEI 99-01 BASIS

a. Provides a description of the rationale for the EAL as provided in NEI 99-01.

2.7.1.10 CNS BASIS

a. Provides CNS-relevant information concerning the EAL.

2.7.1.11 CNS BASIS REFERENCE(S)

a. Site-specific source documentation from which the EAL is derived.

2.8 OPERATING MODE APPLICABILITY

2.8.1 MODES

2.8.1.1 POWER OPERATION

a. Reactor mode switch is in RUN.

2.8.1.2 STARTUP

a. The mode switch is in either REFUEL (with all reactor vessel head closure bolts fully tensioned) or STARTUP/HOT STANDBY.

2.8.1.3 HOT SHUTDOWN

a. The mode switch is in SHUTDOWN with all reactor vessel head closure bolts fully tensioned and reactor coolant temperature is > 212°F.

2.8.1.4 COLD SHUTDOWN

a. The mode switch is in SHUTDOWN with all reactor vessel head closure bolts fully tensioned and reactor coolant temperature is ≤ 212°F.

2.8.1.5 REFUELING

a. The mode switch is in either REFUEL or SHUTDOWN with one or more reactor vessel head closure bolts less than fully tensioned.

2.8.1.6 DEFUELED

a. All reactor fuel removed from reactor pressure vessel (full core off load during refueling or extended outage). This mode is designated as DEF in the EAL Classification Matrix.

2.8.2 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.

2.8.3 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.

2.9 VALIDATION OF INDICATIONS, REPORTS, AND CONDITIONS

2.9.1 All classifications are to be based upon valid indications, reports, or conditions. Indications, reports, or conditions are considered valid when they are verified by (1) an instrument channel check, or (2) indications on related or redundant indications, or (3) by direct observation by plant personnel, such that doubt, related to the indication's operability, the condition's existence, or the reports accuracy is removed. Implicit in this definition is the need for timely assessment.

2.10 PLANNED VS. UNPLANNED EVENTS

2.10.1 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 CNS Technical Specifications. Activities which cause operation beyond that allowed by 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 operating license. However, these conditions may be subject to the reporting requirements of 10CFR50.72.

2.11 CLASSIFYING TRANSIENT EVENTS

2.11.1 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.

2.11.2 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.

2.11.3 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.

2.11.4 Reporting requirements of 10CFR50.72 are applicable and the guidance of NUREG-1022, Event Reporting Guidelines 10CFR50.72 and 50.73, should be applied.

2.12 MULTIPLE SIMULTANEOUS EVENTS AND IMMINENT EAL THRESHOLDS

2.12.1 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; or, an Alert and a Site Area Emergency is a Site Area Emergency. Further guidance is provided in RIS 2007-02, Clarification of NRC Guidance for Emergency Notifications during Quickly Changing Events.

2.12.2 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 classification levels (the early classification may permit more effective implementation of protective measures), it is nonetheless applicable to all emergency classification levels.

2.13 EMERGENCY CLASSIFICATION LEVEL DOWNGRADING

2.13.1 Another important aspect of usable EAL guidance is the consideration of what to do when the risk posed by an emergency is clearly decreasing. A combination approach involving recovery from General Emergencies and some Site Area Emergencies and termination from Unusual Events, Alerts, and certain Site Area Emergencies causing no long term plant damage appears to be the best choice. Downgrading to lower emergency classification levels adds notifications but may have merit under certain circumstances.

2.14 CNS-TO-NEI 99-01 EAL CROSS-REFERENCE

This cross-reference is provided to facilitate association and location of a CNS EAL within the NEI 99-01 IC/EAL identification scheme.

CNS	NEI 99-01, REV 5	
EAL	IC	EXAMPLE EAL
AU1.1	AU1	1
AU1.3	AU1	3
AU2.2	AU2	2
AA1.2	AA1	2
AA2.1	AA2	2
AA3.1	AA3	1
AS1.2	AS1	2
AG1.1	AG1	1
AG1.3	AG1	4

CNS	NEI 99-01, REV 5	
EAL	IC	EXAMPLE EAL
AU1.2	AU1	2
AU2.1	AU2	1
AA1.1	AA1	1
AA1.3	AA1	3
AA2.2	AA2	1
AS1.1	AS1	1
AS1.3	AS1	4
AG1.2	AG1	2

CU1.1	CU3	1
CU2.2	CU2	1
CU3.1	CU4	1
CU4.1	CU6	1, 2
CU6.1	CU7	1
CA2.1	CA1	1, 2
CS2.1	CS1	1
CS2.3	CS1	3
CG2.2	CG1	2

CU2.1	CU1	1
CU2.3	CU2	2
CU3.2	CU4	2
CU5.1	CU8	1
CA1.1	CA3	1
CA3.1	CA4	1, 2
CS2.2	CS1	2
CG2.1	CG1	1

EU1.1	E-HU1	1

FU1.1	FU1	1
FS1.1	FS1	1

FA1.1	FA1	1
FG1.1	FG1	1

CNS	NEI 99-01, REV 5	
EAL	IC	EXAMPLE EAL
HU1.1	HU1	1
HU1.3	HU1	4
HU1.5	HU1	5
HU2.2	HU2	2
HU3.2	HU3	2
HU6.1	HU5	1
HA1.2	HA1	2
HA1.4	HA1	3
HA1.6	HA1	5
HA3.1	HA3	1
HA5.1	HA5	1
HS4.1	HS4	1
HS6.1	HS3	1
HG6.1	HG2	1

CNS	NEI 99-01, REV 5	
EAL	IC	EXAMPLE EAL
HU1.2	HU1	2
HU1.4	HU1	3
HU2.1	HU2	1
HU3.1	HU3	1
HU4.1	HU4	1, 2, 3
HA1.1	HA1	HA1.1
HA1.3	HA1	4
HA1.5	HA1	6
HA2.1	HA2	1
HA4.1	HA4	1, 2
HA6.1	HA6	1
HS5.1	HS2	1
HG4.1	HG1	1, 2

SU1.1	SU1	1
SU3.1	SU2	1
SU5.1	SU4	1
SU6.1	SU5	1, 2
SA2.1	SA2	1
SS1.1	SS1	1
SS4.1	SS6	1
SU8.1	SU6	1, 2
SG2.1	SG2	1

SU2.1	SU8	1
SU4.1	SU3	1
SU5.2	SU4	2
SA1.1	SA5	1
SA4.1	SA4	1
SS2.1	SS2	1
SS7.1	SS3	1
SG1.1	SG1	1

ATTACHMENT 2 EMERGENCY ACTION LEVEL TECHNICAL BASES (INFORMATION USE)

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Off-Site Rad Conditions

Initiating Condition: **Any** release of gaseous or liquid radioactivity to the environment greater than two times the ODAM limits 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 on-going release is detected and the release start time is unknown.

Table A-1 Effluent Monitor Classification Thresholds

	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	ERP	3.50E+08 μCi/sec	3.50E+07 μCi/sec	2.80E+06 μCi/sec	2.24E+05 μCi/sec
GASEOUS	Rx Bldg Vent	3.50E+07 μCi/sec	3.50E+06 μCi/sec	5.45E+05 μCi/sec	8.48E+04 μCi/sec
GASEOUS	Turb Bldg Vent	3.50E+07 μCi/sec	3.50E+06 μCi/sec	5.62E+05 μCi/sec	9.02E+04 μCi/sec
GASEOUS	RW / ARW Bldg Vent	3.50E+07 μCi/sec	3.50E+06 μCi/sec	5.64E+05 μCi/sec	9.08E+04 μCi/sec
LIQUID	Rad Waste Effluent	-----	-----	The lesser of *: 200 x calculated alarm values **OR** monitor upscale	The lesser of *: 2 x calculated alarm values **OR** monitor upscale
LIQUID	Service Water Effluent	-----	-----	4.80E-04 μCi/cc	4.80E-06 μCi/cc

* with effluent discharge **not** isolated

(continued on next page)

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.

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 degradation in these features and/or controls.

The ODAM multiples are specified in AU1.1 and AA1.1 only to distinguish between non-emergency conditions and from each other. While these multiples obviously correspond to an off-site 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 EAL.

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

(continued on next page)

CNS Basis:

The values listed for the Reactor Building vent, Turbine Building vent, and RW/ARW Building vent are based on a value of two times ODAM. The ERP UE value has been set to 1/5 the ODAM instantaneous release limit to provide for a reasonable progression between the UE, Alert, and SAE classifications. If the ERP UE value were placed at two times ODAM, the escalations between the UE, Alert, and SAE levels would have been significantly less than a decade apart. The ERP UE value allows approximately a 1 decade interval between the UE, Alert, and SAE classification points.

Releases in excess of two times the site ODAM (Reference 3) instantaneous limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

CNS Basis Reference(s):

1. Instrument Operating Procedure 4.15, Elevated Release Point and Building Kaman Radiation Monitoring Systems.

2. COR001-18-01, Radiation Monitoring.

3. Off-Site Dose Assessment Manual - ODAM - For Assessment of Gaseous and Liquid Effluents at COOPER NUCLEAR STATION.

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Off-Site Rad Conditions

Initiating Condition: **Any** release of gaseous or liquid radioactivity to the environment greater than two times the ODAM limits for 60 minutes or longer

EAL:

AU1.2 Unusual Event

Any valid liquid effluent 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 on-going release is detected and the release start time is unknown.

Table A-1 Effluent Monitor Classification Thresholds

	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	ERP	3.50E+08 μCi/sec	3.50E+07 μCi/sec	2.80E+06 μCi/sec	2.24E+05 μCi/sec
GASEOUS	Rx Bldg Vent	3.50E+07 μCi/sec	3.50E+06 μCi/sec	5.45E+05 μCi/sec	8.48E+04 μCi/sec
GASEOUS	Turb Bldg Vent	3.50E+07 μCi/sec	3.50E+06 μCi/sec	5.62E+05 μCi/sec	9.02E+04 μCi/sec
GASEOUS	RW / ARW Bldg Vent	3.50E+07 μCi/sec	3.50E+06 μCi/sec	5.64E+05 μCi/sec	9.08E+04 μCi/sec
LIQUID	Rad Waste Effluent	—	—	The lesser of *: 200 x calculated alarm values **OR** monitor upscale	The lesser of *: 2 x calculated alarm values **OR** monitor upscale
LIQUID	Service Water Effluent	—	—	4.80E-04 μCi/cc	4.80E-06 μCi/cc

* with effluent discharge **not** isolated

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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.

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 degradation in these features and/or controls.

The ODAM multiples are specified in AU1.2 and AA1.2 only to distinguish between non-emergency conditions and from each other. While these multiples obviously correspond to an off-site 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 EAL.

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

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CNS Basis:

Liquid releases in excess of two times the site ODAM (Reference 3) instantaneous limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The upper limit for the Radwaste Effluent monitor is currently 0.1 μCi/ml. Therefore, this monitor may not be able to read two times the calculated alarm value. If two times the calculated alarm value exceeds the range of the Radwaste Effluent monitor, the UE condition will be subsumed by the Alert EAL AA1.2. 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.

CNS Basis Reference(s):

1. Instrument Operating Procedure 4.15, Elevated Release Point and Building Kaman Radiation Monitoring Systems.
2. Chemistry Procedure 8.8.11, Liquid Radioactive Waste Discharge Authorization.
3. COR001-18-01, Radiation Monitoring.
4. Off-Site Dose Assessment Manual - ODAM - For Assessment of Gaseous and Liquid Effluents at Cooper Nuclear Station.

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Off-Site Rad Conditions

Initiating Condition: **Any** release of gaseous or liquid radioactivity to the environment greater than two times the ODAM limits 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 ODAM 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 on-going 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.

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 degradation in these features and/or controls.

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The ODAM multiples are specified in AU1.3 and AA1.3 only to distinguish between non-emergency conditions and from each other. While these multiples obviously correspond to an off-site 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 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.).

CNS Basis:

Releases in excess of two times the site Off-Site Dose Assessment Manual (ODAM) (Reference 1) instantaneous limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

CNS Basis Reference(s):

1. Off-Site Dose Assessment Manual - ODAM - For Assessment of Gaseous and Liquid Effluents at Cooper Nuclear Station.

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Off-Site Rad Conditions

Initiating Condition: **Any** release of gaseous or liquid radioactivity to the environment greater than 200 times the ODAM limits 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 on-going release is detected and the release start time is unknown.

Table A-1 Effluent Monitor Classification Thresholds

	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	ERP	3.50E+08 µCi/sec	3.50E+07 µCi/sec	2.80E+06 µCi/sec	2.24E+05 µCi/sec
GASEOUS	Rx Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.45E+05 µCi/sec	8.48E+04 µCi/sec
GASEOUS	Turb Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.62E+05 µCi/sec	9.02E+04 µCi/sec
GASEOUS	RW / ARW Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.64E+05 µCi/sec	9.08E+04 µCi/sec
LIQUID	Rad Waste Effluent	----	----	The lesser of *: 200 x calculated alarm values **OR** monitor upscale	The lesser of *: 2 x calculated alarm values **OR** monitor upscale
LIQUID	Service Water Effluent	----	----	4.80E-04 µCi/cc	4.80E-06 µCi/cc

* with effluent discharge **not** isolated

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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 degradation in these features and/or controls.

The ODAM multiples are specified in AU1.1 and AA1.1 only to distinguish between non-emergency conditions and from each other. While these multiples obviously correspond to an off-site 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.

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

This EAL directly correlates with the IC since annual average meteorology is required to be used in showing compliance with the ODAM and is used in calculating the Table A-1 setpoints.

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

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CNS Basis:

If the Alert thresholds were set at two-hundred times the ODAM limit, the resultant value would exceed the SAE threshold, which is based on 10% of the PAG limits. For this reason, the Alert gaseous thresholds have been set at the log-average of the UE threshold (two times ODAM limit for Turbine Building vent, Radwaste Building vent, and Reactor Building vent, 1/5 the ODAM limit for the ERP) and the SAE threshold (CNS-DOSE dose assessment calculation) (Reference 1). This provides a reasonable escalation in classification from the UE to the Alert and SAE thresholds.

CNS Basis Reference(s):

1. EPIP 5.7.17, CNS-DOSE Assessment, and/or EPIP 5.7.17.1, Dose Assessment (Manual).
2. Instrument Operating Procedure 4.15, Elevated Release Point and Building Kaman Radiation Monitoring Systems.
3. COR001-18-01, Radiation Monitoring.
4. Off-Site Dose Assessment Manual - ODAM - For Assessment of Gaseous and Liquid Effluents at Cooper Nuclear Station.

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Off-Site Rad Conditions

Initiating Condition: **Any** release of gaseous or liquid radioactivity to the environment greater than 200 times the ODAM limits for 15 minutes or longer

EAL:

AA1.2 Alert

Any valid liquid effluent 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 on-going release is detected and the release start time is unknown.

Table A-1 Effluent Monitor Classification Thresholds

	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	ERP	3.50E+08 μCi/sec	3.50E+07 μCi/sec	2.80E+06 μCi/sec	2.24E+05 μCi/sec
GASEOUS	Rx Bldg Vent	3.50E+07 μCi/sec	3.50E+06 μCi/sec	5.45E+05 μCi/sec	8.48E+04 μCi/sec
GASEOUS	Turb Bldg Vent	3.50E+07 μCi/sec	3.50E+06 μCi/sec	5.62E+05 μCi/sec	9.02E+04 μCi/sec
GASEOUS	RW / ARW Bldg Vent	3.50E+07 μCi/sec	3.50E+06 μCi/sec	5.64E+05 μCi/sec	9.08E+04 μCi/sec
LIQUID	Rad Waste Effluent	----	----	The lesser of *: 200 x calculated alarm values **OR** monitor upscale	The lesser of *: 2 x calculated alarm values **OR** monitor upscale
LIQUID	Service Water Effluent	----	----	4.80E-04 μCi/cc	4.80E-06 μCi/cc

* with effluent discharge **not** isolated

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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 degradation in these features and/or controls.

The ODAM multiples are specified in AU1.2 and AA1.2 only to distinguish between non-emergency conditions and from each other. While these multiples obviously correspond to an off-site 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 Radwaste Effluent radiation monitor readings to exceed the threshold identified in the IC established by the radioactivity discharge permit.

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

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CNS Basis:

This event escalates from the Unusual Event by escalating the magnitude of the release by a factor of 100. The Radwaste Effluent monitor may not be able to read two-hundred times the calculated alarm value because the top of its scale is 0.1 μCi/ml. Two-hundred times the calculated alarm value is to be assumed if the Radwaste Effluent monitor reading goes off scale high while discharges are in-progress via this path.

CNS Basis Reference(s):

1. Instrument Operating Procedure 4.15, Elevated Release Point and Building Kaman Radiation Monitoring Systems.
2. Chemistry Procedure 8.8.11, Liquid Radioactive Waste Discharge Authorization.
3. COR001-18-01, Radiation Monitoring.
4. Off-Site Dose Assessment Manual - ODAM - For Assessment of Gaseous and Liquid Effluents at Cooper Nuclear Station.

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Off-Site Rad Conditions

Initiating Condition: **Any release** of gaseous or liquid radioactivity to the environment greater than 200 times the ODAM limits for 15 minutes or longer

EAL:

AA1.3 Alert

Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODAM 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 on-going 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.

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 degradation in these features and/or controls.

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The ODAM multiples are specified in AU1.3 and AA1.3 only to distinguish between non-emergency conditions and from each other. While these multiples obviously correspond to an off-site 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.).

CNS Basis:

None

CNS Basis Reference(s):

1. Off-Site Dose Assessment Manual - ODAM - For Assessment of Gaseous and Liquid Effluents at Cooper Nuclear Station.

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Off-Site Rad Conditions

Initiating Condition: Off-site dose resulting from an actual or imminent release of gaseous radioactivity greater than 0.1 Rem TEDE or 0.5 Rem thyroid CDE for the actual or projected duration of the release

EAL:

AS1.1 Site Area Emergency

Any valid gaseous monitor 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.

Table A-1 Effluent Monitor Classification Thresholds

	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	ERP	3.50E+08 μCi/sec	3.50E+07 μCi/sec	2.80E+06 μCi/sec	2.24E+05 μCi/sec
GASEOUS	Rx Bldg Vent	3.50E+07 μCi/sec	3.50E+06 μCi/sec	5.45E+05 μCi/sec	8.48E+04 μCi/sec
GASEOUS	Turb Bldg Vent	3.50E+07 μCi/sec	3.50E+06 μCi/sec	5.62E+05 μCi/sec	9.02E+04 μCi/sec
GASEOUS	RW / ARW Bldg Vent	3.50E+07 μCi/sec	3.50E+06 μCi/sec	5.64E+05 μCi/sec	9.08E+04 μCi/sec
LIQUID	Rad Waste Effluent	----	----	The lesser of *: 200 x calculated alarm values **OR** monitor upscale	The lesser of *: 2 x calculated alarm values **OR** monitor upscale
LIQUID	Service Water Effluent	----	----	4.80E-04 μCi/cc	4.80E-06 μCi/cc

* with effluent discharge **not** isolated

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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 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.

CNS Basis:

The Table A-1 Site Area Emergency thresholds have been determined using CNS-DOSE dose projection calculations (Reference 1). The Site Area Emergency effluent monitor readings are one decade less than the General Emergency values.

For the purposes of this EAL, the Site Boundary for CNS is a 1 mile radius around the plant (Reference 1).

CNS Basis Reference(s):

1. EPIP 5.7.17, CNS-DOSE Assessment, and/or EPIP 5.7.17.1, Dose Assessment (Manual).
2. CNS Drawing DWG.2.2 (P3-A-45).

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Off-Site Rad Conditions

Initiating Condition: Off-site dose resulting from an actual or imminent release of gaseous radioactivity greater than 0.1 Rem TEDE or 0.5 Rem thyroid CDE for the actual or projected duration of the release

EAL:

AS1.2 Site Area Emergency Dose assessment using actual meteorology indicates doses > 0.1 Rem TEDE or > 0.5 Rem 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 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.

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CNS Basis:

The dose rate EALs are based on a Site Boundary dose rate of 0.1 Rem/hr TEDE or 0.5 Rem/hr CDE thyroid, whichever is more limiting. Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible.

For the purposes of this EAL, the Site Boundary for CNS is a 1 mile radius around the plant.

CNS Basis Reference(s):

1. EPIP 5.7.17, CNS-DOSE Assessment, and/or EPIP 5.7.17.1, Dose Assessment (Manual).

2. CNS-DOSE.

3. CNS Drawing DWG.2.2 (P3-A-45).

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Off-Site Rad Conditions

Initiating Condition: Off-site dose resulting from an actual or imminent release of gaseous radioactivity greater than 0.1 Rem TEDE or 0.5 Rem thyroid CDE for the actual or projected duration of the release

EAL:

AS1.3 Site Area Emergency

Field survey indicates closed window dose rates > 0.1 Rem/hr that is expected to continue for ≥ 60 min. at or beyond the site boundary (NOTE 1)

OR

Field survey sample analysis indicates thyroid CDE > 0.5 Rem for 1 hr of inhalation at or beyond the site boundary

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.

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.

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CNS Basis:

The 0.5 Rem integrated CDE thyroid dose was established in consideration of the 1:5 ratio of the EPA Protective Action Guidelines for TEDE and thyroid exposure. In establishing the field survey emergency action levels, a duration of 1 hour is assumed. Therefore, the dose rate EALs are based on a Site Boundary dose rate of 0.1 Rem/hr TEDE or 0.5 Rem for 1 hour of inhalation CDE thyroid, whichever is more limiting.

For the purposes of this EAL, the Site Boundary for CNS is a 1 mile radius around the plant.

CNS Basis Reference(s):

1. CNS Drawing DWG.2.2 (P3-A-45).

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Off-Site Rad Conditions

Initiating Condition: Off-site dose resulting from an actual or imminent release of gaseous radioactivity greater than 1 Rem TEDE or 5 Rem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

AG1.1 General Emergency

Any valid gaseous monitor 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.

Table A-1 Effluent Monitor Classification Thresholds

	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	ERP	3.50E+08 µCi/sec	3.50E+07 µCi/sec	2.80E+06 µCi/sec	2.24E+05 µCi/sec
	Rx Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.45E+05 µCi/sec	8.48E+04 µCi/sec
	Turb Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.62E+05 µCi/sec	9.02E+04 µCi/sec
	RW / ARW Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.64E+05 µCi/sec	9.08E+04 µCi/sec
LIQUID	Rad Waste Effluent	—	—	The lesser of *: 200 x calculated alarm values **OR** monitor upscale	The lesser of *: 2 x calculated alarm values **OR** monitor upscale
	Service Water Effluent	—	—	4.80E-04 µCi/cc	4.80E-06 µCi/cc

* with effluent discharge **not** isolated

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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 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.

CNS Basis:

The Table A-1 General Emergency thresholds have been determined using CNS-DOSE dose projection calculations (Reference 1). The General Emergency effluent monitor readings are one decade greater than the Site Area Emergency values.

For the purposes of this EAL, the Site Boundary for CNS is a 1 mile radius around the plant (Reference 2).

CNS Basis Reference(s):

1. EPIP 5.7.17, CNS-DOSE Assessment, and/or EPIP 5.7.17.1, Dose Assessment (Manual).
2. CNS Drawing DWG.2.2 (P3-A-45).
3. Computer Dose Projection (CNS DOSE).

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Off-Site Rad Conditions

Initiating Condition: Off-site dose resulting from an actual or imminent release of gaseous radioactivity greater than 1 Rem TEDE or 5 Rem 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 Rem TEDE or > 5 Rem 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 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.

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CNS Basis:

The General Emergency values are based on the boundary dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 1 Rem TEDE or 5 Rem CDE thyroid for the actual or projected duration of the release. Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible.

For the purposes of this EAL, the Site Boundary for CNS is a 1 mile radius around the plant.

CNS Basis Reference(s):

1. EPIP 5.7.17, CNS-DOSE Assessment, and/or EPIP 5.7.17.1, Dose Assessment (Manual).
2. CNS DOSE.
3. CNS Drawing DWG.2.2 (P3-A-45).

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 1 - Off-Site Rad Conditions

Initiating Condition: Off-site dose resulting from an actual or imminent release of gaseous radioactivity greater than 1 Rem TEDE or 5 Rem 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 Rem/hr expected to continue for ≥ 60 min. at or beyond the site boundary (NOTE 1)

OR

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

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.

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.

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CNS Basis:

The 5 Rem integrated CDE thyroid dose was established in consideration of the 1:5 ratio of the EPA Protective Action Guidelines for TEDE and thyroid exposure. In establishing the dose rate emergency action levels, a duration of 1 hour is assumed. Therefore, the dose rate EALs are based on a Site Boundary dose rate of 1 Rem/hr TEDE or 5 Rem for 1 hour of inhalation CDE thyroid, whichever is more limiting.

For the purposes of this EAL, the Site Boundary for CNS is a 1 mile radius around the plant.

CNS Basis Reference(s):

1. CNS Drawing DWG.2.2 (P3-A-45).

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 2 - On-Site Rad Conditions And Spent Fuel Pool Events

Initiating Condition: Unplanned rise in plant radiation levels

EAL:

AU2.1 Unusual Event

Unplanned water level drop in the reactor cavity or spent fuel pool as indicated by **any** of the following:

- LI-86 (calibrated to 1001' elev.)
- Spent fuel pool low level alarm
- Visual observation

AND

Valid Area radiation monitor reading rise on RMA-RA-1 or RMA-RA-2

Mode Applicability:

All

NEI 99-01 Basis:

This EAL addresses increased radiation levels as a result of water level decreases above irradiated fuel. 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 combination of refueling cavity and spent fuel pool. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.

For example, a refueling bridge ARM reading may increase due to planned evolutions such as head lift or even a fuel assembly being raised in the manipulator mast. Also, a monitor could in fact be properly responding to a known event involving transfer or relocation of a source, stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Generally, increased radiation monitor indications will need to combined with another indicator (or personnel report) of water loss.

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For refueling events where the water level drops below the RPV flange classification would be via CU2.1. This event escalates to an Alert per 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.

CNS Basis:

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

The major item of concern on loss of inventory from the reactor cavity and SFP is to maintain adequate water level for personnel shielding and cooling of the irradiated fuel. Normal SFP water level is 37' 6 1/2" above the bottom. A low SFP level alarm can be determined by Annunciator 9-4-2/A-3, FUEL POOL COOLING TROUBLE, alarming due to Annunciator Panel 25-15, Fuel Pool Low Level at 4" below normal. Decreases in SFP water level can also be detected through visual observation. The Skimmer Surge Tank low level alarm (Annunciator 9-4-2/C-3 at 100 ft^3 in the skimmer surge tank, elevation 981' 3") alone may not be conclusive evidence of an uncontrolled loss of inventory from the SFP. SFP weir wall design should prevent inadvertent draining of the SFP through Fuel Pool Cooling and Demineralizer System connections. A Skimmer Surge Tank low level alarm needs to be confirmed by visual observation to determine the extent of inventory loss from the SFP (Reference 1, 3, 4).

During refueling when the RPV head is removed, Shutdown Range RPV water level instrument NBI-LI-86 is recalibrated to read vessel cavity level up to the 1001' elevation (Refuel Floor). With the reactor cavity in communication with the Spent Fuel Pool via the fuel transfer canal, uncontrolled inventory loss can be remotely monitored via this indicator. NBI-LI-86 can be used only if it has been set up to read to 1001' elevation as specified in Procedure 4.6.1, Reactor Vessel Water Level Indication (Reference 5).

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Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment. Technical Specification Section LCO 3.7.6 (Reference 7) requires spent fuel storage pool water level be maintained at least 21' 6" over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks. Technical Specification LCO 3.9.6 (Reference 8) requires RPV water level to be maintained at least 21' above the top of the RPV flange. During refueling, this maintains sufficient water level in the refueling cavity and SFP to retain iodine fission product activity in the water in the event of a fuel handling accident.

Area radiation monitors that may indicate a loss of shielding of spent fuel in the SFP or refueling cavity include (Reference 2, 6):

- RMA-RA-1 (1448) RX BLDG FUEL POOL (HR) AREA.
- RMA-RA-2 (1449) RX BLDG FUEL POOL (LR) AREA.

Portable radiation monitors are routinely employed to conduct radiation surveys in the Reactor Building. This source of information should not be excluded when considering emergency classification under this EAL, particularly when RMA-RA-1 and RMS-RA-2 may be taken out of service for preventative or corrective maintenance.

CNS Basis Reference(s):

1. System Operating Procedure 2.2.32, Fuel Pool Cooling and Demineralizer System.
2. Alarm Procedure 2.3_9-3-1, Panel 9-3 - Annunciator 9-3-1.
3. Alarm Procedure 2.3_9-4-2, Panel 9-4 - Annunciator 9-4-2, C-3.
4. Abnormal Procedure 2.4FPC, Fuel Pool Cooling Trouble.
5. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
6. Emergency Procedure 5.1RAD, Building Radiation Trouble.
7. Technical Specification LCO 3.7.6.
8. Technical Specification LCO 3.9.6.

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 2 - On-Site Rad Conditions And 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 1,000 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, 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.

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 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.

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CNS Basis:

It is recognized that some plant area radiation monitors may not be able to detect or display a reading that is one-thousand times NORMAL LEVELS. The intent of this IC is to rely on currently installed plant monitors and not to require design changes/back fits. In cases where an installed area radiation monitor cannot detect or display values at or above 1,000 X NORMAL LEVELS value, then survey instrument results may be used.

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 Control Room annunciator and local alarms will be energized to warn of abnormal or significantly changing radiological conditions. The alarm limit is normally set at approximately ten times normal background for each channel (Reference 1, 2).

Routine and work specific surveys are conducted throughout the station at frequencies specified by RP Management. Routine surveys are scheduled per the RP Department surveillance schedule. Work specific surveys are conducted in accordance with the Radiation Work Permit (RWP) (Reference 3).

CNS Basis Reference(s):

1. Alarm Procedure 2.3_9-3-1, Panel 9-3 - Annunciator 9-3-1.
2. Emergency Procedure 5.1RAD, Building Radiation Trouble.
3. Rad Protection Procedure 9.ALARA.4, Radiation Work Permits.

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 2 - On-Site Rad Conditions And 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 **EITHER** of the following:

Valid RMA-RA-1 Fuel Pool Area Rad reading > 50 R/hr

OR

Valid RMP-RM-452 A-D Rx Bldg Vent Exhaust Plenum Hi-Hi alarm

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.

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Escalation of this emergency classification level, if appropriate, would be based on EALs in Subcategory A1.

CNS 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 refueling cavity or Spent Fuel Pool (SFP).

The bases for the ventilation radiation Hi-Hi alarm is a spent fuel handling accident (Reference 2). Fuel Pool area radiation > 50 R/hr represents 100 times the high alarm setpoint (HR) and is unambiguously indicative of spent fuel damage or uncovery (Reference 1).

CNS Basis Reference(s):

1. Alarm Procedure 2.3_9-3-1, Panel 9-3 - Annunciator 9-3-1, Alarm A-10.
2. Alarm Procedure 2.3_9-4-1, Panel 9-4 - Annunciator 9-4-1, Alarm E-4.

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 2 - On-Site Rad Conditions And 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 refueling 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 EALs in Subcategory A1.

CNS 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).

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The major item of concern on loss of inventory from the Spent Fuel Pool and Refueling Cavity is to maintain adequate water level for personnel shielding and cooling of the irradiated fuel. Normal Spent Fuel Pool water level is 37' 6 1/2" above the bottom. A low pool level alarm occurs at 4" below the normal water level. Decreases in Spent Fuel Pool water level can also be detected only through visual observation and the existence of the Skimmer Surge Tank low level alarm (9-4-2/C-3) at 100 ft^3 in the skimmer surge tank which is at elevation 981' 3" (Reference 1, 2, 3).

During refueling when the RPV head is removed, Shutdown Range RPV water level instrument NBI-LI-86 is recalibrated to read vessel cavity level up to the 1001' elevation (Refuel Floor). With the reactor cavity in communication with the Spent Fuel Pool via the fuel transfer canal, uncontrolled inventory loss can be remotely monitored via this indicator. NBI-LI-86 can be used only if it has been set up to read to 1001' elevation as specified in Procedure 4.6.1, Reactor Vessel Water Level Indication (Reference 4).

Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment. Technical Specification Section LCO 3.7.6 (Reference 5) requires spent fuel storage pool water level be maintained at least 21' 6" over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks. Technical Specification LCO 3.9.6 (Reference 6) requires RPV water level to be maintained at least 21' above the top of the RPV flange. During refueling, this maintains sufficient water level in the refueling cavity and SFP to retain iodine fission product activity in the water in the event of a fuel handling accident.

CNS Basis Reference(s):

1. System Operating Procedure 2.2.32, Fuel Pool Cooling and Demineralizer System.
2. Annunciator Procedure 2.3_ 9-4-2, Panel 9-4 - Annunciator 9-4-2.
3. Abnormal Procedure 2.4FPC, Fuel Pool Cooling Trouble.
4. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
5. Technical Specification LCO 3.7.6.
6. Technical Specification LCO 3.9.6.

Category: A - Abnormal Rad Release/Rad Effluent

Subcategory: 2 - On-Site Rad Conditions And Spent Fuel Pool Events

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 mRem/hr in **EITHER** of the following areas requiring continuous occupancy to maintain plant safety functions:

Main Control Room (RM-RA-20)

OR

CAS

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 EAL may be involved.

The value of 15 mRem/hr is derived from the GDC 19 value of 5 Rem in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements", provides that the 15 mRem/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an Alert.

Areas requiring continuous occupancy include the Main Control Room and the Central Alarm Station (CAS).

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CNS Basis:

Areas that meet this threshold include the Main Control Room and the Central Alarm Station. The Central Alarm Station is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations.

There are no permanently installed CAS Area radiation monitors that may be used to assess this EAL threshold. Therefore, this portion of the EAL threshold must be assessed using local radiation survey (Reference 1, 2).

CNS Basis Reference(s):

1. Alarm Procedure 2.3_9-3-1, Panel 9-3 - Annunciator 9-3-1, B-10.
2. Emergency Procedure 5.1RAD, Building Radiation Trouble.

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 1 - Loss of Power

Initiating Condition: AC power capability to critical 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 critical buses

EAL:

CU1.1 Unusual Event

AC power capability to critical 4160V Buses 1F and 1G 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 critical 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 Station Service Transformer • Emergency Station Service Transformer • Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer (Note 8)
Onsite
• DG-1 • DG-2

NOTE 8 – The time required to establish the backfeed is likely longer than the specified time interval. If off-normal plant conditions have already established the backfeed, its power to the safety-related buses may be considered an off-site power source.

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Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

NEI 99-01 Basis:

The condition indicated by this EAL is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a loss of all AC power to critical buses. This condition could occur due to a loss of off-site power with a concurrent failure of all but one emergency generator to supply power to its emergency bus. The subsequent loss of this single power source would escalate the event to an Alert in accordance with CA1.1.

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

CNS Basis:

The 4160V Critical Buses 1F (Div 1) and 1G (Div 2) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following off-site sources of power: Figure C-1 illustrates the 4160V AC Distribution System (Reference 7, 8).

- Startup Transformer - The Startup Transformer provides a source of off-site AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of off-site AC power to the station whenever the main generator is off-line. The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the station Normal Transformer fails or that the main generator trips off-line.
- Emergency Transformer - The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by the 69 kV transmission line from OPPD. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.

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- Backfeeding power from the 345 kV line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on-line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. As mentioned in NOTE 8, the time required to establish the backfeed is likely longer than the 15 minute interval. If off-normal plant conditions have already established the backfeed its power to the safety-related buses may be considered an off-site power source.

On-site power sources are the emergency diesel generators (DG-1 and DG-2).

If critical bus AC power is reduced to a single source for > 15 minutes, an Unusual Event is declared under this EAL.

If a power supply should have picked up a critical bus but failed to do so, that power supply should be considered unavailable until it has been successfully tied onto the bus. If the power supply can be tied to the bus within 15 minutes, then the power supply is to be considered available.

The Supplemental Diesel Generator (SDG) is not considered an on-site or an off-site emergency power supply and is not considered in classifications involving loss of power. The SBO Coping Time per Regulatory Guide 1.155 considers the impact of a SDG.

This cold condition EAL is equivalent to the hot condition loss of AC power EAL SA1.1.

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CNS Basis Reference(s):

1. System Operating Procedure 2.2.15, Startup Transformer.
2. System Operating Procedure 2.2.16, Normal Station Service Transformer.
3. System Operating Procedure 2.2.17, Emergency Station Service Transformer.
4. System Operating Procedure 2.2.18, 4160V Auxiliary Power Distribution System.
5. System Operating Procedure 2.2.20, Standby AC Power System (Diesel Generator).
6. Emergency Procedure 5.3SBO, Station Blackout.
7. BR 3001, One Line Diagram.
8. BR 3002, Sheet 1.
9. NC43456, One Line Switching Diagram 161kV Substation.
10. Enercon Services, Inc. Report No. NPP1-PR-01, Station Blackout Coping Assessment for Cooper Nuclear Station, Revision 2.

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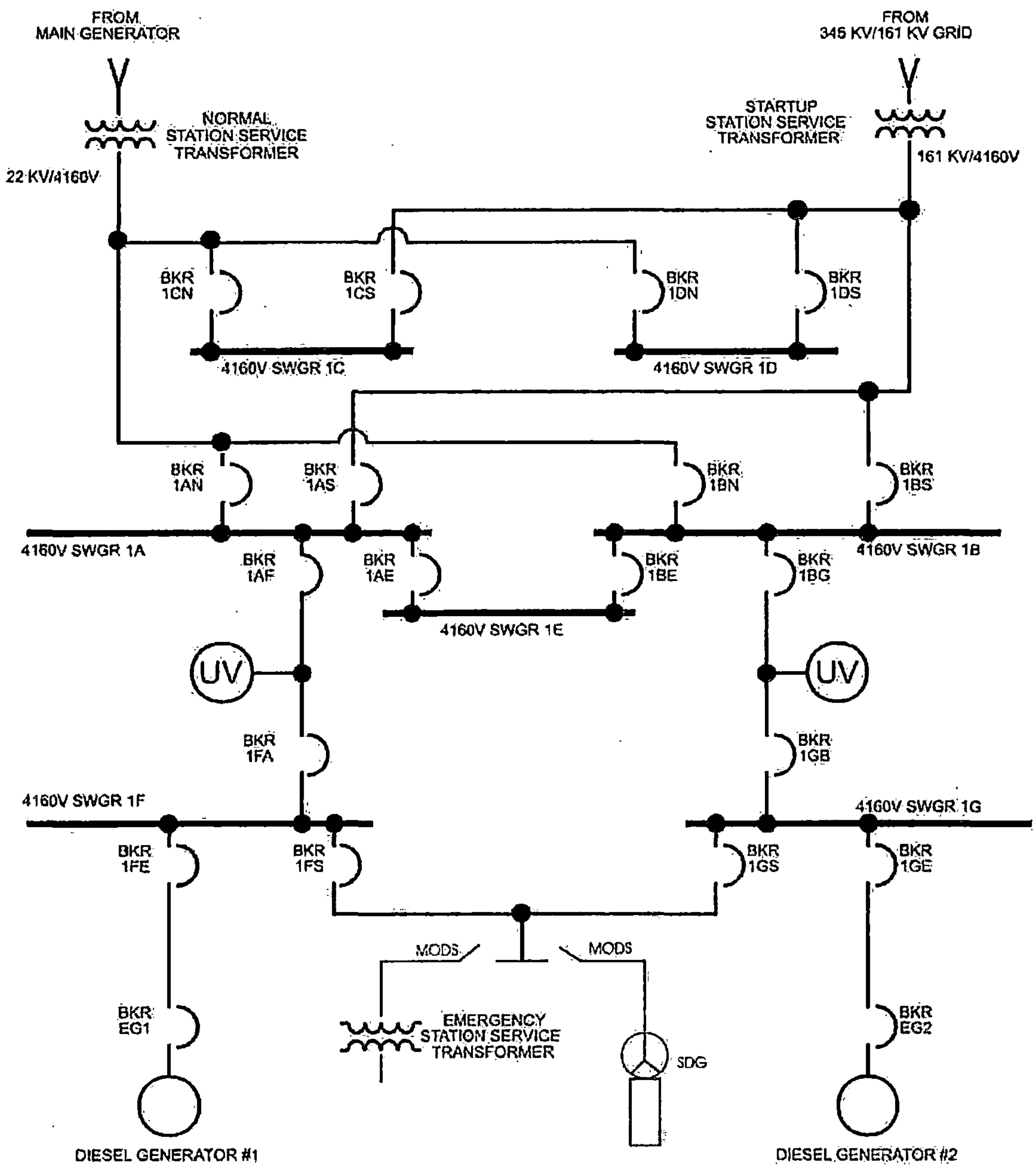
Figure C-1: 4160V AC Distribution System



Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 1 - Loss of Power

Initiating Condition: Loss of **all** off-site and **all** on-site AC power to critical buses for 15 minutes or longer

EAL:

CA1.1 Alert

Loss of **all** off-site and **all** on-site AC power (Table C-4) to critical 4160V Buses 1F and 1G 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 Station Service Transformer • Emergency Station Service Transformer • Backfeed 345 kv line through Main Power Transformer to the Normal Station Service Transformer (Note 8)
Onsite
• DG-1 • DG-2

NOTE 8 – The time required to establish the backfeed is likely longer than the specified time interval. If off-normal plant conditions have already established the backfeed, its power to the safety-related buses may be considered an off-site power source.

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Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, 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 critical buses, relative to that specified for the Site Area Emergency EAL.

Escalating to Site Area Emergency, if appropriate, is by Abnormal Rad Levels/Radiological Effluent EALs.

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

CNS Basis:

The 4160V Critical Buses 1F (Div 1) and 1G (Div 2) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following off-site sources of power: Figure C-1 illustrates the 4160V AC Distribution System (Reference 7, 8).

- Startup Transformer - The Startup Transformer provides a source of off-site AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of off-site AC power to the station whenever the main generator is off-line (< 160 MWe). The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the station Normal Transformer fails or that the main generator trips off-line.

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- Emergency Transformer - The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by the 69 kV transmission line from OPPD. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.
- Backfeeding power from the 345 kV line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. As mentioned in NOTE 8, the time required to establish the backfeed is likely longer than the 15 minute interval. If off-normal plant conditions have already established the backfeed its power to the safety-related buses may be considered an off-site power source.

On-site power sources are the emergency diesel generators (DG-1 and DG-2).

If a power supply should have picked up a critical bus but failed to do so, that power supply should be considered unavailable until it has been successfully tied onto the bus. If the power supply can be tied to the bus within 15 minutes, then the power supply is to be considered available.

The Supplemental Diesel Generator (SDG) is not considered an on-site or an off-site emergency power supply and is not considered in classifications involving loss of power. The SBO Coping Time per Regulatory Guide 1.155 considers the impact of a SDG.

This EAL is the cold condition equivalent of the hot condition loss of all AC power EAL SS1.1. When in Cold Shutdown, Refueling, or Defueled Mode, the event can be classified as an Alert because of the significantly reduced decay heat, lower temperature and pressure, increasing the time to restore one of the emergency buses, relative to that existing when in hot conditions.

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CNS Basis Reference(s):

1. System Operating Procedure 2.2.15, Startup Transformer.
2. System Operating Procedure 2.2.16, Normal Station Service Transformer.
3. System Operating Procedure 2.2.17, Emergency Station Service Transformer.
4. System Operating Procedure 2.2.18, 4160V Auxiliary Power Distribution System.
5. System Operating Procedure 2.2.20, Standby AC Power System (Diesel Generator).
6. Emergency Procedure 5.3SBO, Station Blackout.
7. BR 3001, One Line Diagram.
8. BR 3002, Sheet 1.
9. NC43456, One Line Switching Diagram 161kV Substation.
10. Enercon Services, Inc. Report NPP1-PR-01, Station Blackout Coping Assessment for Cooper Nuclear Station, Revision 2.

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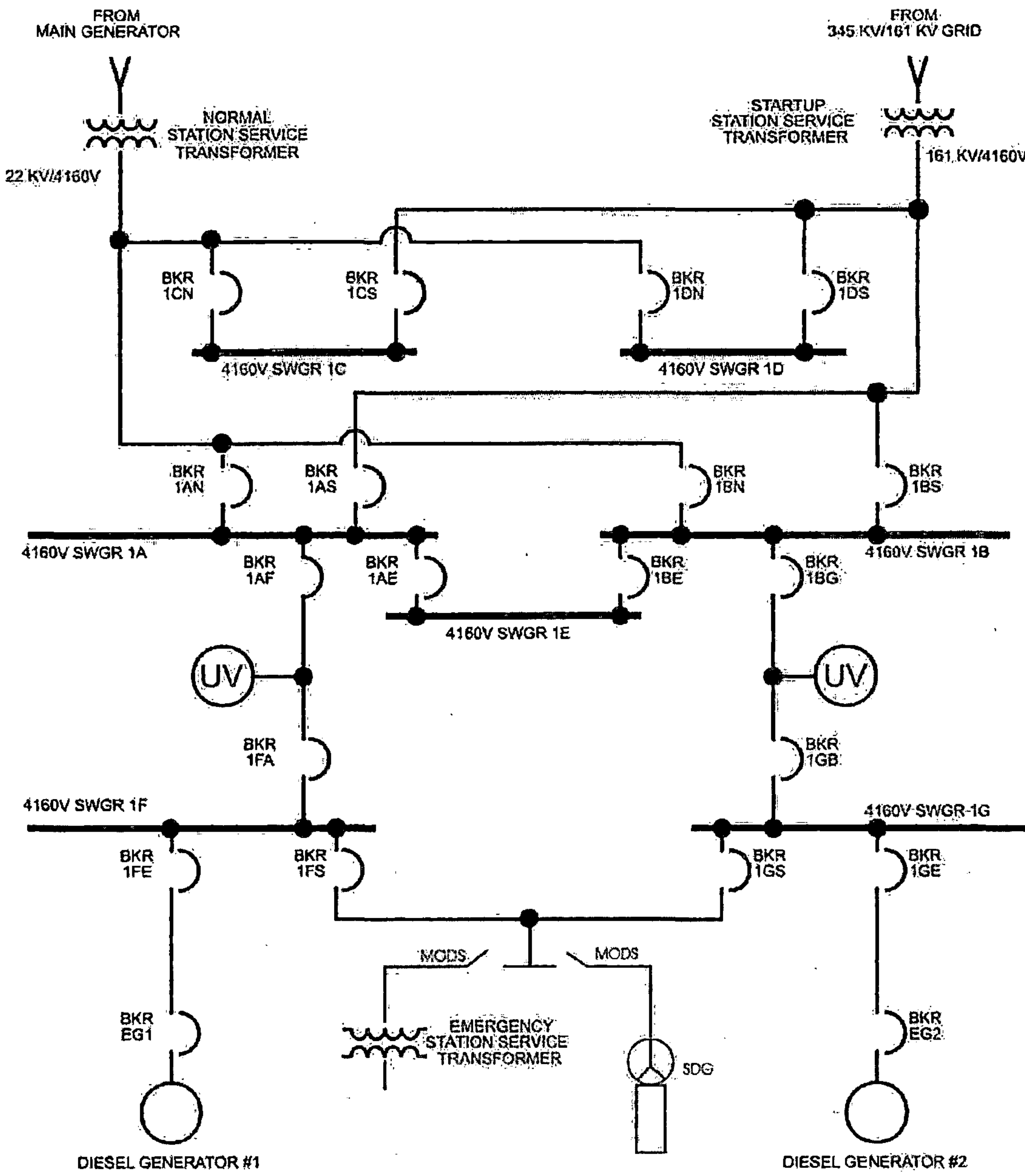
Figure C-1: 4160V AC Distribution System



Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 2 - RPV Level

Initiating Condition: Unplanned loss of RPV inventory

EAL:

CU2.1 Unusual Event

RPV level **cannot** be restored and maintained > +3 in. for ≥ 15 min. (NOTE 3) due to RCS leakage

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 per 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.

The difference between CU2.1 and CU2.2 deals with the RCS conditions that exist between Cold Shutdown and Refueling Modes. In Cold Shutdown, the RPV will normally be intact and RPV level is typically controlled below the elevation of the RPV flange and above the low-end of the normal control band. In the Refueling Mode the RPV is not intact and any planned evolutions to lower RPV level below the elevation of the RPV flange must be carefully controlled.

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CNS 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 +3 in. (low level scram setpoint), level is well below the normal control band and automatic RPS and PCIS actuations are required (Reference 3, 5).

RPV level is normally monitored using the following instruments (Reference 1, 2):

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.).
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.).
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.).
- Narrow Range RFC-LI-94A, B & C (0 to 60 in.).
- Shutdown Range NBI-LI-86 (0 to 400 in.).

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

This Cold Shutdown EAL represents the hot condition EAL SU6.1, in which RCS leakage is associated with Technical Specification limits. In Cold Shutdown, these limits are not applicable; hence, the use of RPV level as the parameter of concern in this EAL (Reference 4).

CNS Basis Reference(s):

1. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
2. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
3. EOP-1A RPV Control.
4. NEI/NRC EAL FAQ #2006-014.
5. Technical Specification Table 3.3.1.1-1.

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 (LI-86: 206 in. normal calibration, 113.75 in. elevated calibration)

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 - Refueling

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 an Unusual Event due to the reduced RCS inventory that is available to keep the core covered.

The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame, then it may indicate a more serious condition exists.

(continued on next page)

Continued loss of RCS inventory will result in escalation to the Alert emergency classification level via either CA2.1 or CA3.1.

The difference between CU2.1 and CU2.2 deals with the RPV conditions that exist between Cold Shutdown and Refuel Modes. In Cold Shutdown, the RCS will normally be intact and standard RPV inventory and level monitoring means are available. In the Refuel Mode, the RCS is not intact and RPV level and inventory may be monitored by different means.

This EAL involves a decrease in RPV level below the top of the RPV flange or a decrease below the RPV level band (when the RPV level band is established below 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 AU2.1, until such time as the level decreases to the level of the vessel flange.

If RPV level continues to decrease and reaches the Low-Low ECCS actuation setpoint, then escalation to CA2.1 would be appropriate.

CNS Basis:

The RPV flange is 722.75 in. above the RPV bottom head. RPV water level at this elevation is normally indicated by the Shutdown Range instrument (LI-86 Shutdown, 0 - 400 in.). When calibrated for normal plant operations, the Shutdown Range instrument reads 206 in. at the RPV flange. With the RPV head removed, the instrument is calibrated to indicate reactor cavity water levels as high as the refuel floor. When calibrated for elevated indication, the Shutdown Range instrument reads 113.75 in. at the RPV flange. Visual observation of water level in the reactor cavity and RPV is also used during refuel operations.

CNS Basis Reference(s):

1. General Operating Procedure 2.1.20.3, RPV Refueling Preparation (Wet Lift or Dryer and Separator), Indicated RPV Level vs. RPV Height attachment.
2. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
3. IAC Procedure 14.15.3, Reactor Vessel Open Head Level Monitor System.

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 water level rise • RPV make-up rate rise • Observation of unisolable RCS leakage

Mode Applicability:

5 - Refueling

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.

Continued loss of RCS Inventory will result in escalation to the Alert emergency classification level via either CA2.1 or CA3.1.

(continued on next page)

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 level changes listed in Table C-1. Sump 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.

Escalation to the Alert emergency classification level would be via either CA2.1 or CA3.1.

CNS Basis:

RPV level is normally monitored using the following instruments (Reference 3, 4):

- Wide Range NBI-LI-85A, B & C (-155 to 60 in.).
- Steam Nozzle Range NBI-LI-92 (0 to 180 in.).
- Fuel Zone Range NBI-LI-91A, B & C (-320 to 60 in.).
- Narrow Range RFC-LI-94A, B & C (0 to 60 in.).
- Shutdown Range NBI-LI-86 (0 to 400 in.).

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs.

(continued on next page)

In this EAL, all water level indication is unavailable and the RPV inventory loss should 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 (Reference 1). 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 water level could be indicative of RHR valve misalignment or leakage (Reference 2). If the make-up 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.

CNS Basis Reference(s):

1. System Operating Procedure 2.2.27, Equipment, Floor, and Chemical Drain System.
2. System Operating Procedure 2.2.69, Residual Heat Removal System.
3. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
4. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.

Category: C - Cold Shutdown/Refueling System Malfunction

Subcategory: 2 - RPV Level

Initiating Condition: Loss of RPV inventory

EAL:

CA2.1 Alert

RPV level < -42 in.

OR

RPV level **cannot** be monitored for ≥ 15 min. (NOTE 3) with **any** unexplained RPV leakage indication, Table C-1

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 • Suppression pool water level rise • RPV make-up rate rise • Observation of unisolable RCS leakage

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

(continued on next page)

Enclosure 2

Emergency Plan Implementing Procedure 5.7.8, Revision 27

COOPER NUCLEAR STATION

Operations Manual

Emergency Preparedness

EMERGENCY PLAN IMPLEMENTING PROCEDURE

5.7.8

ACTIVATION OF OSC

Level of Use:	**MULTIPLE**
Quality:	**QAPD RELATED**
Effective Date:	**3/28/16**
Approval Authority:	**ITR-RDM**
Procedure Owner:	**EMERGENCY PREP ON-SITE COORD**

TABLE OF CONTENTS

1. ENTRY CONDITIONS (INFORMATION USE)

   1.1 OSC activation required per Emergency Director.

2. INSTRUCTIONS (REFERENCE USE)

   2.1 **PERFORM** Attachment 1 to activate and operate OSC.

ATTACHMENT 1 OSC ACTIVATION AND OPERATION (REFERENCE USE)

NOTE – OSC required to be activated within ~ 1 hour from time of declaration of ALERT or higher, or from time when ED directs ERO activation at NOUE.

1. ACTIVATION OF OSC

1.1 **ASSUME** position of OSC Supervisor.

1.1.1 **CARD** into Personnel Accountability Reader at OSC door.

1.1.2 **SIGN IN** on staffing board.

1.1.3 **OBTAIN** copy of EPIPs and Emergency Telephone Directory.

1.1.4 **ENSURE** communications (telephone, gaitronics, alternate intercom) established with TSC.

1.2 **ENSURE** minimum staffing positions are met and ready to activate OSC.

- Radiation Protection Technicians (6 minimum).
- Mechanics (2 minimum).
- Electricians (2 minimum).
- Instrument and Control Technicians (2 minimum).

1.2.1 IF necessary to meet minimum OSC staffing, THEN **PERFORM** interim staffing as follows:

1.2.1.1 **ASSIGN** individual to fill vacant position with skills necessary to perform function of position.

1.2.1.2 **RECORD** names and positions in facility log.

1.3 **ANNOUNCE** OSC is activated.

1.4 **INFORM** Maintenance Coordinator in TSC that OSC is activated.

1.5 **RECORD** time of activation in facility log.

2. OPERATION OF OSC

2.1 **ENSURE** equipment repair and restoration priorities established by TSC are being followed:

2.1.1 **CONFERENCE** with TSC and **UPDATE** priorities as required.

2.2 **VERIFY** OSC Technical Communicator in communications with Control Room.

NOTE – Goal of OSC is to assemble and dispatch Repair or Survey Teams into field within 20 minutes.

2.3 **PROVIDE** support and **COORDINATE** Repair/Survey Teams.

2.3.1 **MAINTAIN** TSC Maintenance Coordinator informed of team mission and status of each team.

2.3.2 **COORDINATE** material and equipment needs with Logistic Coordinator, as applicable.

2.3.3 **DETERMINE** primary fix and contingency fix.

2.3.4 **SELECT** appropriate OSC Lead based on mission of team.

2.3.5 **DIRECT** OSC Lead(s) to form Repair/Survey Team(s).

2.4 **CONDUCT** periodic OSC briefs following significant change in plant status or approximately every 30 to 60 minutes.

2.4.1 **USE** Attachment 2 as necessary.

2.5 **PERFORM** shift turnover as required.

NOTE – Conditional actions in Table 1 apply continuously during performance of Section 2.

3. CONDITIONAL ACTIONS FOR OSC

3.1 IF condition in Table 1, Column 1, met, THEN **PERFORM** actions in Column 2.

Table 1

(1) IF/WHEN condition met:		(2) THEN **PERFORM** following:
1	OSC becomes uninhabitable or unsafe, or loses ability to function.	1. **EVACUATE** normal OSC in Admin Bldg. 2. **PERFORM** EPIP 5.7.8.1 to establish Alternate OSC in IAC Shop (Turbine Bldg. 932').©[2]
2	Alternate OSC cannot be activated or must be relocated off-site.	1. **EVACUATE** Alternate OSC. 2. **PERFORM** EPIP 5.7.8.2 to establish Alternate Off-Site OSC in EOF.
3	Plant conditions change significantly with repair teams dispatched.	1. **DIRECT** Chemistry/RP Lead to assess impact of plant conditions on dispatched Repair Teams. 2. **DETERMINE** if recall of dispatched teams required.
4	Additional personnel or materials needed.	**DIRECT** Logistics Coordinator to obtain additional personnel or materials, as needed.
5	All OSC ERO positions filled.	**DISMISS** extra ERO personnel to West Warehouse.
6	Event terminated.	**GO TO** Step 4 to return OSC to readiness.

4. RETURN TO READINESS

4.1 **PERFORM** following to return OSC to readiness:

- EP Workstations returned to standby.
- EP documents returned to storage location.
- Emergency response equipment returned to standby.
- OSC Team dispatch forms, records, and notes provided to EP Coordinator.

4.2 **RELEASE** OSC personnel to attend brief.

ATTACHMENT 2 OSC BRIEFING GUIDE (INFORMATION USE)

1. **PROVIDE** notice prior to facility brief, if time allows.

   Example: "Attention in the OSC. Brief in 5 minutes."

2. **ANNOUNCE** the brief.

   Example: "Attention in the OSC for a Brief."

3. **ENSURE** personnel are attentive.

4. If appropriate, **REVIEW** current emergency action level (EAL) classification declared with its basis.

5. **REVIEW** following:
   - Priority of each OSC Repair Team.
   - Repair Team status.
   - Plant status.

6. **RECORD** time in facility log.

ATTACHMENT 3 INFORMATION SHEET (INFORMATION USE)

1. PURPOSE©[1]

1.1 Provides guidance for activation and operation of Operations Support Center (OSC) in event of an ALERT or higher classification. This procedure is responsibility of OSC Supervisor.

1.2 ERO is not normally activated per Emergency Plan at NOUE but may be activated if additional assistance is required to respond to emergency. If ERO activation occurs at NOUE, then procedure applies.

2. PRECAUTIONS AND LIMITATIONS

2.1 Interim staffing of OSC Supervisor shall be approved by Emergency Director. Interim staffing of remaining OSC Staff shall be approved by OSC Supervisor.

3. DISCUSSION

3.1 OSC is designated assembly area for initial accountability for OSC Staff.

3.2 OSC is assembly and staging area for CNS personnel for emergency response assignments.

3.3 OSC is located on 903' elevation of Administration Building near TSC.

3.4 OSC provides a location where plant logistic support can be coordinated during an emergency.

3.5 When fully manned, OSC is staffed with following personnel:

3.5.1 Welders.

3.5.2 Machinists.

3.5.3 Mechanics.

3.5.4 Electrical shop personnel.

3.5.5 IAC Shop personnel.

3.5.6 Utility Shop personnel.

3.5.7 Warehouse personnel.

3.5.8 Technical Communicator.

3.5.9 Radiological Protection personnel.

3.5.10 Clerical personnel.

3.6 Repair, rescue, and Radiological Monitoring Team members are chosen from OSC Staff by OSC Lead personnel who in their opinion are best suited for a particular team mission.

3.7 OSC Leaders shall brief team members on task assignment.

3.8 Positional Instruction Manuals (PIMs) are numbered and controlled by Emergency Preparedness Department, labeled by ERO position, and are located in OSC.

3.8.1 Chemistry/Radiological Protection Lead - PIM #2.

3.8.2 Mechanical Lead - PIM #3.

3.8.3 Electrical Lead - PIM #4.

3.8.4 I&C Lead - PIM #5.

3.8.5 Utility Lead - PIM #6.

3.8.6 Warehouse Personnel - PIM #7.

3.8.7 OSC Clerk - PIM #8.

3.8.8 Technical Communicator - PIM CR#1.

4. RECORDS

4.1 No quality records are generated by this procedure.

5. REFERENCES

5.1 CODES AND STANDARDS

5.1.1 NPPD Emergency Plan for CNS.

5.1.2 NUREG 0654, Revision 1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants.

5.2 PROCEDURES

5.2.1 Administrative Procedure 0-EP-01, Emergency Response Organization Responsibilities.

5.2.2 Emergency Plan Implementing Procedure 5.7COMMUN, Communications.

5.2.3 Emergency Plan Implementing Procedure 5.7.1, Emergency Classification.

5.2.4 Emergency Plan Implementing Procedure 5.7.8.1, Activation of Alternate OSC.

5.2.5 Emergency Plan Implementing Procedure 5.7.8.2, Activation of Alternate Off-Site OSC/TSC.

5.2.6 Emergency Plan Implementing Procedure 5.7.15, OSC Team Dispatch.

5.2.7 Emergency Plan Implementing Procedure 5.7.21, Maintaining Emergency Preparedness - Emergency Exercises, Drills, Tests, and Evaluations.

5.3 MISCELLANEOUS

5.3.1 RCR 2002-0126.

5.4 NRC COMMITMENTS

5.4.1 ©[1] NRC Commitment 811217-01-08. Response to IR 81-13, Develop and Implement Procedures for Activation of Emergency Facilities. Commitment affects entire procedure (Attachment 3, Step 1 flagged).

5.4.2 ©[2] NRC Commitment NSD921204-01, Response to IR 92-14, Provide for an Alternate Operations Support Center (OSC) in the Event Personnel Contamination Prevents Return of Teams to Primary OSC. Commitment affects Attachment 1, Table 1, Line 1.

Enclosure 3

Emergency Plan Implementing Procedure 5.7.16, Revision 25

COOPER NUCLEAR STATION

Operations Manual

Emergency Preparedness

EMERGENCY PLAN IMPLEMENTING PROCEDURE

5.7.16

RELEASE RATE DETERMINATION

Level of Use:	**MULTIPLE**
Quality:	**QAPD RELATED**
Effective Date:	**3/28/16**
Approval Authority:	**ITR-RDM**
Procedure Owner:	**EP MANAGER**

TABLE OF CONTENTS

1. ENTRY CONDITION (INFORMATION USE)

    1.1 Radiological release determination required with CNS-DOSE unavailable.

2. INSTRUCTIONS (REFERENCE USE)

    2.1 **SELECT** and **PERFORM** appropriate release rate determination attachment per Table 1.

Table 1

DETERMINATION	CONDITION	METHOD USING	GO TO
ERP or Ground Level Noble Gas Release Rate	• KAMAN(s) available.	• ERP KAMAN • Reactor Bldg KAMAN • Turbine Bldg KAMAN • RW/ARW Bldg KAMAN	Attachment 1
	• ERP KAMAN Noble Gas not available. • MSIVs OPEN.	Steam Jet Air Ejectors (SJAEs)	Attachment 2 and Attachment 3
	All of following met: • ERP KAMAN Noble Gas not available. • Primary Containment (PC) intact. • No PC vent occurring or anticipated.	In-Containment radiation monitors	Attachment 4 and Attachment 5
	All of following met: • ERP KAMAN Noble Gas not available. • PC venting occurring or anticipated, either controlled or uncontrolled.	Drywell Curie Content and either actual or estimated Vent Flow Rate	Attachment 6 and Attachment 7
Liquid Release	**PERFORM** when estimate of total curies released off-site required.	Sampling and Curie Content Calculations	Attachment 11

    2.2 IF any of following conditions met:
    - Changes of ± 20% in KAMAN effluent monitor readings when above NOUE.
    - Minimum of every hour for effective ages 0 to 10 hours.
    - Every 10 hours for effective ages 10 to 100 hours.

    THEN **RECALCULATE** release rate.

ATTACHMENT 1 EFFLUENT RELEASE RATE READING AVAILABLE ON KAMAN DISPLAY HARD CARD (REFERENCE USE)

CAUTION – Readings on KAMAN monitors will be erroneous if respective exhaust fans are not running.

1. **VERIFY** exhaust fans are operating for selected KAMAN monitor.
2. **DETERMINE** release rate from one of following:
   - PMIS Display PMIS05
   - Table 1-1 PMIS Data Point below.

Table 1-1

PMIS DATA POINT	ASSOCIATED KAMAN MONITOR
N8000	Reactor Building KAMAN Effluent Average Flow
N8003	Turbine Building KAMAN Effluent Average Flow
N8007	AOG and Radwaste Building KAMAN Effluent Average Flow
N8012	ERP KAMAN Effluent Average Flow

3. **ENSURE** KAMAN monitor is displaying release rate in units of micro curies per second (μCi/sec), Parameter Number 49.

4. IF normal mode of KAMAN process flow not available,
   THEN **REFER** to Procedures 4.15.1, 4.15.2, 4.15.3, and 4.15.4, as applicable, to operate selected KAMAN in MONITOR OPERATION WITHOUT PROCESS FLOW PARAMETER using flow estimation.

5. Concurrently **PERFORM** EPIP 5.7.17.1.

   5.1 **INPUT** release rate value in μCi/sec into EPIP 5.7.17.1 in support of dose assessment.

6. **REPEAT** above steps for follow-up release rate determination and dose assessment.

ATTACHMENT 2 RELEASE RATE USING STEAM JET AIR EJECTOR MONITORS (REFERENCE USE)

1. **VERIFY** MSIVs open.

NOTE – Reactor shutdown is met by any of following conditions:
- All rods fully inserted.
- Reactor scram and power below 3%.
- Hot shutdown boron weight injected per EOPs.

2. **DETERMINE** time since shutdown (EFFECTIVE AGE) in hours.

   2.1 **OBTAIN** or **VERIFY** time of reactor shutdown from Radiological Control Manager (RCM), Operations, and/or Engineering.

   2.2 **RECORD** EFFECTIVE AGE in hours in Table 3-1, Column 1.

      2.2.1 IF not shutdown,
      THEN **RECORD** zero (0).

3. **OBTAIN** highest SJAE monitor reading in mrem/hr from any of following:

   3.1 OFF-GAS RAD Monitors PNL 9-02 behind front panel in Control Room.

   3.2 PMIS Display PMIS05.

   3.3 PMIS Data Points.
   - SJAE A: N082.
   - SJAE B: N083.

   3.4 **RECORD** SJAE MONITOR (mrem/hr) reading in Table 3-1, Column 2.

NOTE – Conversion factor will change as fuel degrades. Chemistry is responsible for updating conversion factor after sampling.

   3.5 **OBTAIN** current SJAE conversion factor (mrem/hr to μCi/sec/cfm) from one of following:
   - Posted location on PNL 9-02 near OFF GAS RAD Monitor behind front panel in Control Room.
   - PMIS Data Point SPDS0215.
   - Chemistry.

   3.6 **RECORD** CONVERSION FACTOR value in Table 3-1, Column 3.

4. **OBTAIN** combined SJAE flow rate (cfm) from one of following:
   - OFF-GAS RAD Monitor in Control Room.
   - PMIS Display PMIS05.
   - PMIS Data Points:
     - SJAE A: N084
     - SJAE B: N085

5. **RECORD** COMBINED SJAE FLOW RATE (cfm) value in Table 3-1, Column 4.

6. **CALCULATE** Noble Gas Release Rate on Table 3-1 as follows:

   6.1 **MULTIPLY** Column 2 by Column 3 by Column 4.

   (Column 2) x (Column 3) x (Column 4) = (Column 5)

   6.2 **RECORD** NOBLE GAS RELEASE RATE (μCi/sec) results in Table 3-1, Column 5.

7. Concurrently **PERFORM** EPIP 5.7.17.1.

   7.1 **INPUT** Table 3-1, Column 5, NOBLE GAS RELEASE RATE value in μCi/sec into EPIP 5.7.17.1 in support of dose assessment.

8. **REPEAT** above steps for follow-up release rate determination and dose assessment.

ATTACHMENT 3 RELEASE RATE USING STEAM JET AIR EJECTOR MONITORS DATA SHEET (INFORMATION USE)

Table 3-1

(1) EFFECTIVE AGE (hours)	(2) SJAE MONITOR (mrem/hr)	(3) CONVERSION FACTOR (mrem/hr to μCi/sec/cfm)	(4) COMBINED SJAE FLOW RATE (cfm)	(5) NOBLE GAS RELEASE RATE (μCi/sec) (2) x (3) x (4)

Completed By: ______________________________ Date: ____________

ATTACHMENT 4 ERP RELEASE RATE DETERMINATION USING PRIMARY CONTAINMENT MONITORS (REFERENCE USE)

NOTE – Reactor shutdown is met by any of following conditions:

- All rods fully inserted.
- Reactor scram and power below 3%.
- Hot shutdown boron weight injected per EOPs.

1. **DETERMINE** time since shutdown (EFFECTIVE AGE) in hours.

   1.1 **OBTAIN** or **VERIFY** time of reactor shutdown from RCM, Operations, and/or Engineering.

   1.2 **RECORD** EFFECTIVE AGE in hours in Table 5-1, Column 1.

      1.2.1 IF not shutdown,
      THEN **RECORD** zero (0).

2. **OBTAIN** highest exposure rate of high range in-containment radiation monitor in rem/hr from one of following:

   - In-containment monitor readouts located in Control Room behind front panel labeled PNL 9-02 DRYWELL RAD MONITOR RMA-RM-40A & RMA-RM-40B.
   - PMIS Display PMIS05 or SPDS01.

   2.1 **RECORD** ACTUAL CONTAINMENT EXPOSURE RATE (rem/hr) value in Table 5-1, Column 2.

3. **USE** EFFECTIVE AGE and **DETERMINE** projected DBA-LOCA exposure rate (rem/hr) from Attachment 8.

   3.1 **RECORD** PROJECTED DBA-LOCA EXPOSURE RATE AT EFFECTIVE AGE (rem/hr) value in Table 5-1, Column 3.

4. **USE** EFFECTIVE AGE to determine projected DBA-LOCA noble gas release rate (Ci/sec) from Attachment 9.

   4.1 **RECORD** PROJECTED DBA-LOCA NOBLE GAS RELEASE RATE (Ci/sec) value in Table 5-1, Column 4.

5. **CALCULATE** Noble Gas Release Rate on Table 5-1 as follows:

   5.1 **DIVIDE** Column 2 by Column 3 to determine fraction of DBA-LOCA that has occurred.

   5.2 **MULTIPLY** results by Column 4 by Column 5.

   $$\frac{\text{(Column 2)}}{\text{(Column 3)}} \times \text{(Column 4)} \times \text{(Column 5)} = \text{(Column 6)}$$

   5.3 **RECORD** NOBLE GAS RELEASE RATE (μCi/sec) value in Table 5-1, Column 6.

6. Concurrently **PERFORM** EPIP 5.7.17.1.

   6.1 **INPUT** Table 5-1, Column 6, NOBLE GAS RELEASE RATE value in μCi/sec into EPIP 5.7.17.1 in support of dose assessment.

7. **REPEAT** above steps for follow-up release rate determination and dose assessment.

ATTACHMENT 5 ERP RELEASE RATE DETERMINATION USING PRIMARY CONTAINMENT MONITORS DATA SHEET (INFORMATION USE)

Table 5-1

(1) EFFECTIVE AGE (hours)	(2) ACTUAL CONTAINMENT EXPOSURE RATE (rem/hr)	(3) PROJECTED DBA-LOCA EXPOSURE RATE AT EFFECTIVE AGE (rem/hr) (from Att. 8)	(4) PROJECTED DBA-LOCA NOBLE GAS RELEASE RATE (Ci/sec) (from Att. 9)	(5) CONVERSION FACTOR (Ci to μCi)	(6) NOBLE GAS RELEASE RATE (μCi/sec) $\frac{(2)}{(3)} \times (4) \times (5)$
				1E6	
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Completed By: ______________________________ Date: ____________

ATTACHMENT 6 RELEASE RATE DETERMINATION BASED ON DRYWELL CURIE CONTENT AND VENT FLOW RATE (REFERENCE USE)

NOTE – Reactor shutdown is met by any of following conditions:
- All rods fully inserted.
- Reactor scram and power below 3%.
- Hot shutdown boron weight injected per EOPs.

1. **DETERMINE** time since shutdown (EFFECTIVE AGE) in hours.

   1.1 **OBTAIN** or **VERIFY** time of reactor shutdown from RCM, Operations, and/or Engineering.

   1.2 **RECORD** time in hours in Table 7-1, Column 1, EFFECTIVE AGE.

      1.2.1 IF not shutdown,
      THEN **RECORD** zero (0).

2. **OBTAIN** highest exposure rate of high range in-containment radiation monitor in rem/hr from one of following:

   - In-containment monitor readouts located in Control Room behind front panel labeled PNL 9-02 DRYWELL RAD MONITOR RMA-RM-40A & RMA-RM-40B.
   - PMIS Display PMIS05 or SPDS01.

   2.1 **RECORD** ACTUAL CONT. EXPOSURE RATE value in rem/hr in Table 7-1, Column 2.

3. **USE** EFFECTIVE AGE to determine projected DBA-LOCA dose rate from Attachment 8.

   3.1 **RECORD** PROJECTED DBA-LOCA EXPOSURE RATE AT EFFECTIVE AGE value in rem/hr in Table 7-1, Column 3.

CAUTION – Error in release rate determination will occur if either IODINE or TOTAL curves are used on Attachment 10.

4. **APPLY** EFFECTIVE AGE to NOBLE GAS curve and **DETERMINE** projected DBA-LOCA noble gas drywell curie content from Attachment 10.

   4.1 **RECORD** PROJECTED DBA-LOCA NOBLE GAS CURIE CONTENT (Ci) value in Table 7-1, Column 4.

5. **CALCULATE** Estimated Drywell Noble Gas Curie Content on Table 7-1.

   5.1 **DIVIDE** Column 2 by Column 3 to determine fraction of DBA-LOCA.

   5.2 **MULTIPLY** results by Column 4.

   $$\frac{\text{(Column 2)}}{\text{(Column 3)}} \times \text{(Column 4)} = \text{(Column 5)}$$

   5.3 **RECORD** ESTIMATED DRYWELL NOBLE GAS CURIE CONTENT (Ci) value in Table 7-1, Column 5.

6. **DETERMINE** Noble Gas Concentration (Ci/ft^3) in drywell on Table 7-1.

   6.1 **DIVIDE** Column 5 by Column 6.

   (Column 5) ÷ (Column 6) = (Column 7)

   6.2 **RECORD** NOBLE GAS CONCENTRATION (Ci/ft^3) value in Table 7-1, Column 7.

7. **USE** Table 6-1 to determine DRYWELL VENTING FLOW RATE.

> **NOTE** – Exact vent flow rates for containment pressures may not be available, resulting in using calculated maximum vent flow rate in Table 6-1.

   7.1 **SELECT** Maximum Vent Flow rate based on vent flow path and Containment Pressure.

7.2 IF venting from both Drywell and Torus PC lines,
THEN **USE** vent flow rate of 638 cfm based on calculated maximum flow rate per line in Table 6-1.

Table 6-1

VENTING FROM:	ASSUMED CONTAINMENT PRESSURE	CALCULATED MAXIMUM VENT FLOW RATE
Torus via PC-MO-1308 and PC-MO-305	65 psid	319 cfm
Drywell via PC-MO-1310 and PC-MO-306	65 psid	319 cfm
Torus via Hard Pipe Vent	53.8 > P ≥ 65 psid	12,067 cfm
	P ≤ 53.8 psid	12,002 cfm

7.3 **RECORD** DRYWELL VENTING FLOW RATE (cfm) value in Table 7-1, Column 8.

8. **CALCULATE** NOBLE GAS RELEASE RATE on Table 7-1.

8.1 **MULTIPLY** Column 7 by Column 8 by Column 9.

(Column 7) x (Column 8) x (Column 9) = (Column 10)

8.2 **RECORD** NOBLE GAS RELEASE RATE (μCi/sec) value in Table 7-1, Column 10.

9. Concurrently **PERFORM** EPIP 5.7.17.1.

9.1 **INPUT** Table 7-1, Column 10, NOBLE GAS RELEASE RATE value in μCi/sec into EPIP 5.7.17.1 to determine dose assessment.

10. **REPEAT** above steps for follow-up release rate determination and dose assessment.

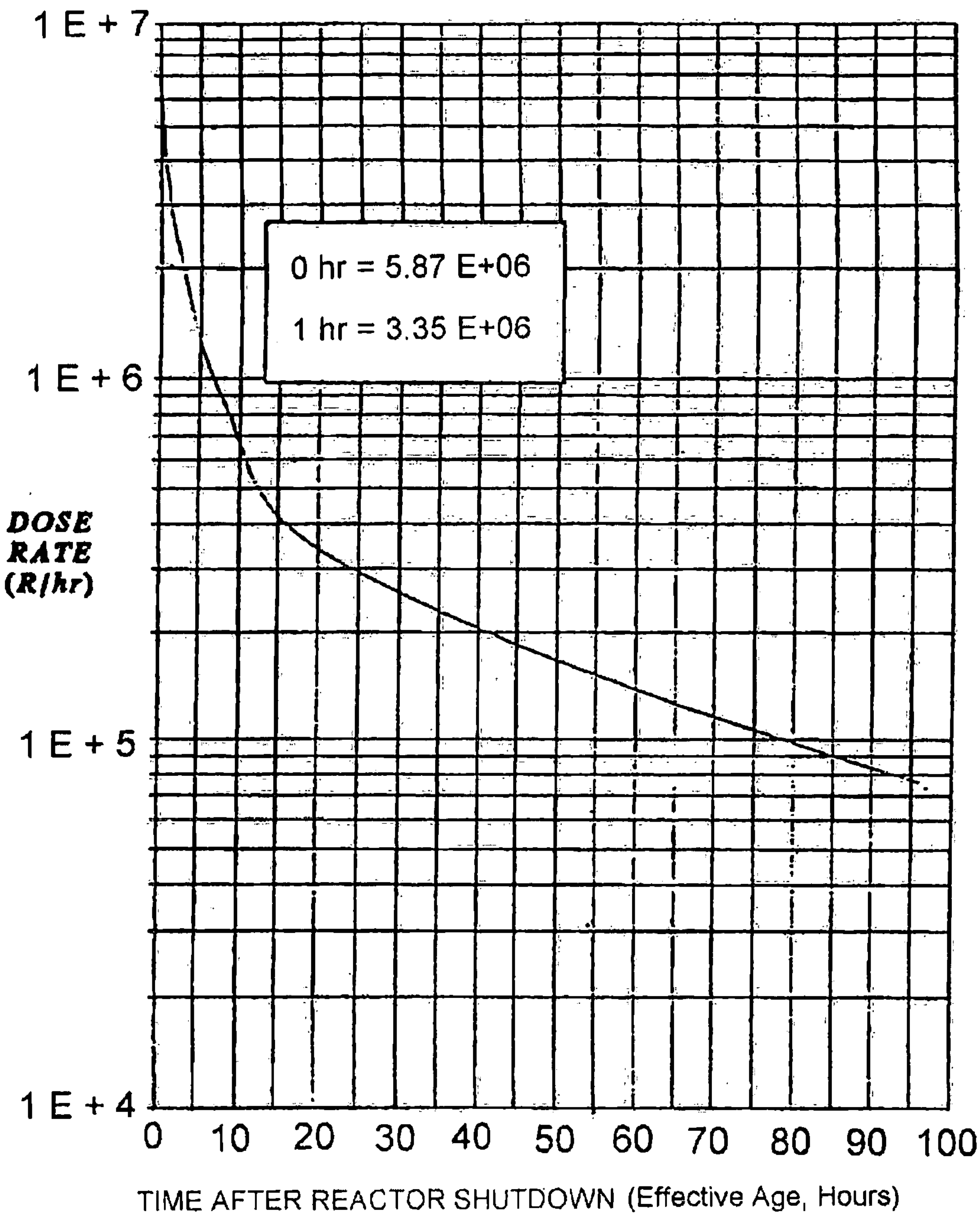
ATTACHMENT 7 RELEASE RATE DETERMINATION BASED ON DRYWELL CURIE CONTENT AND VENT FLOW RATE DATA SHEET (INFORMATION USE)

Table 7-1

(1) EFFECTIVE AGE (hours)	(2) ACTUAL CONT. EXPOSURE RATE (rem/hr)	(3) PROJECTED DBA-LOCA EXPOSURE RATE AT EFFECTIVE AGE (rem/hr) (from Att. 8)	(4) PROJECTED DBA-LOCA NOBLE GAS CURIE CONTENT (Ci) (from Att. 10)	(5) ESTIMATED DRYWELL NOBLE GAS CURIE CONTENT (Ci) $\frac{(2)}{(3)} \times (4)$	(6) D/W VOLUME (ft^3)	(7) NOBLE GAS CONC. (Ci/ft^3) (5) ÷ (6)	(8) DRYWELL VENTING FLOW RATE (cfm)	(9) CONV. FACTOR (Ci/min to µCi/sec)	(10) NOBLE GAS RELEASE RATE (µCi/sec) (7) x (8) x (9)
					1.45E5			1.67E4	
					1.45E5			1.67E4	
					1.45E5			1.67E4	
					1.45E5			1.67E4	
					1.45E5			1.67E4	
					1.45E5			1.67E4	
					1.45E5			1.67E4	
					1.45E5			1.67E4	
					1.45E5			1.67E4	
					1.45E5			1.67E4	
					1.45E5			1.67E4	
					1.45E5			1.67E4	

Completed By: ______________________________ Date: ________

ATTACHMENT 8 PROJECTED DRYWELL DOSE RATE VS. TIME FOR DBA-LOCA (INFORMATION USE)



5.7.16A.TIF

Figure 8-1

ATTACHMENT 9 PROJECTED NOBLE GAS RELEASE RATE AT ERP VS. TIME FOR DBA-LOCA (INFORMATION USE)

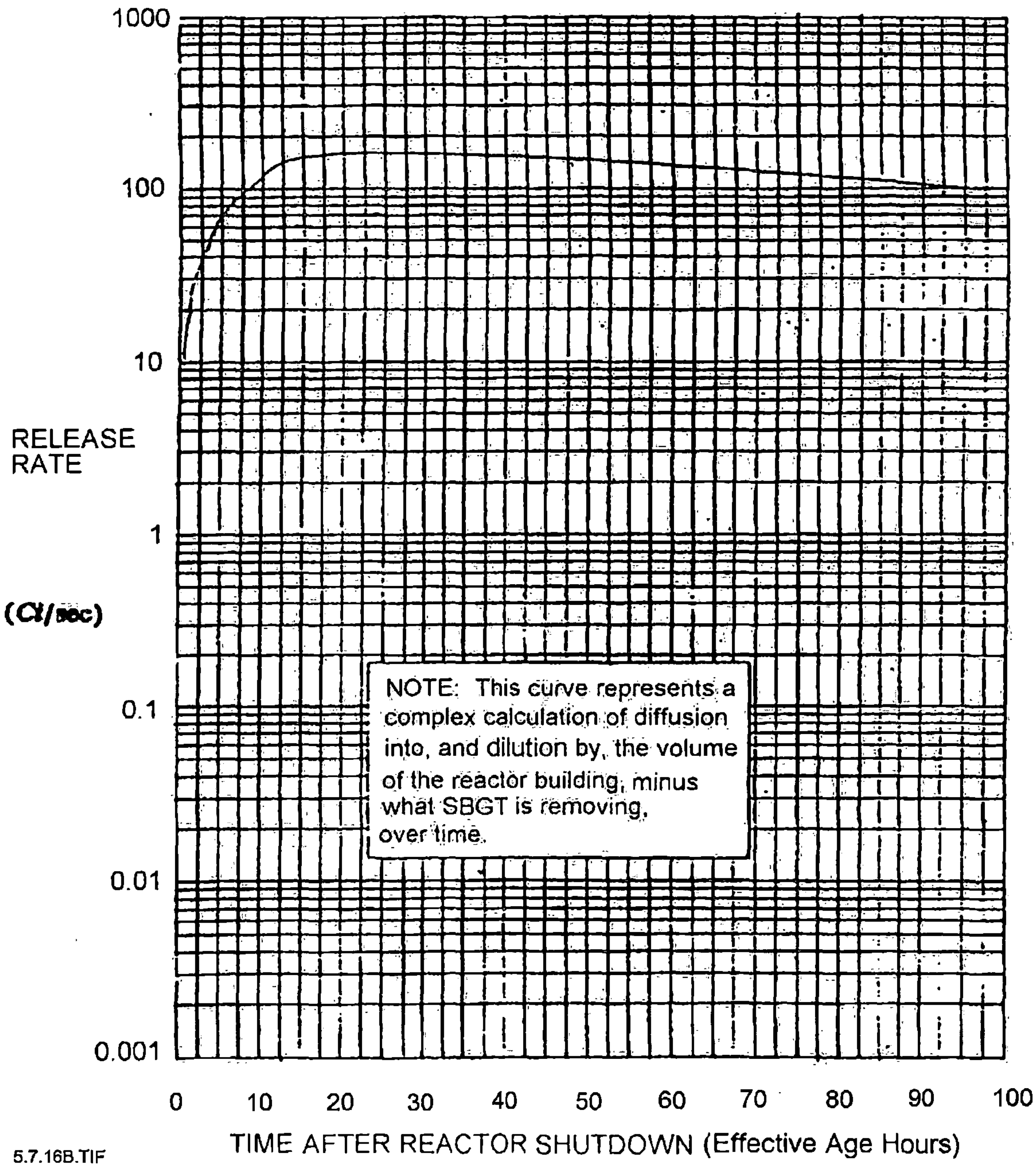


Figure 9-1

ATTACHMENT 10 PROJECTED DRYWELL CURIE CONTENT VS. EFFECTIVE AGE FOR DBA-LOCA (INFORMATION USE)

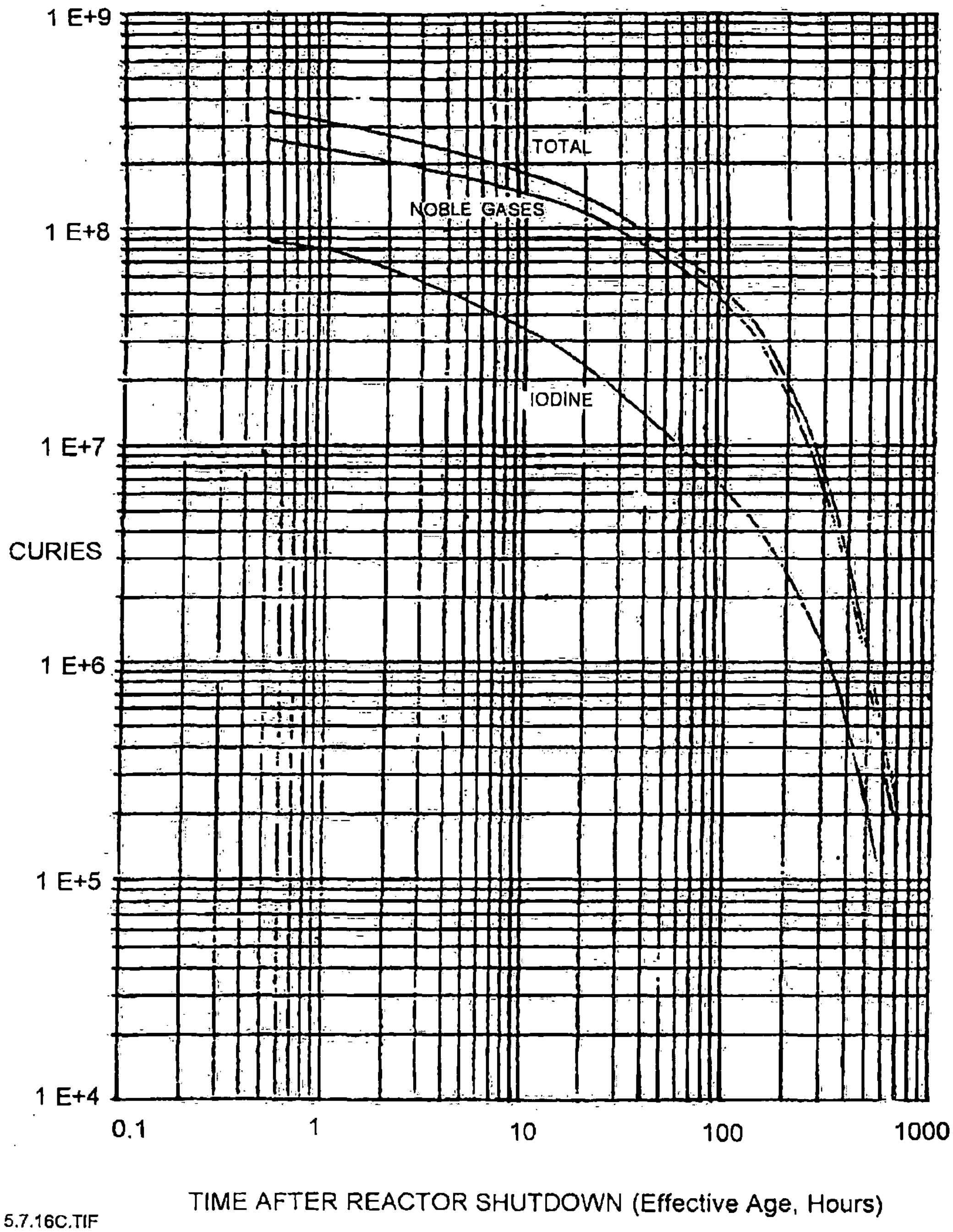


Figure 10-1

ATTACHMENT 11 LIQUID RELEASE CURIE CONTENT CALCULATION (REFERENCE USE)

1. **USE** Table 11-1 to determine appropriated section for selected tank to determine volume released.

Table 11-1	
SELECTED TANK	GO TO
South Condensate Storage Tank (CST-A)	Step 2
North Condensate Storage Tank (CST-B)	Step 3
Other Tanks with Potentially Contaminated Liquids	Step 4

2. **USE** Table 12-1 to determine volume released from South Condensate Storage Tank (CST-A).

   2.1 **OBTAIN** CST-A level (ft) prior to release from one of following:

   - Control Room Indicator CM-LIC-5, COND STOR TK 1A LEVEL at Panel A.
   - PMIS Data Point F001.

   2.1.1 IF CST-A level prior to release cannot be determined, THEN **ASSUME** CST-A was at maximum operational level of 36 feet.

   2.1.2 **RECORD** CST-A Initial Level (ft) value in Table 12-1, Column 1.

   2.2 **OBTAIN** CST-A level (ft) following release from one of following:

   - Control Room Indicator CM-LIC-5, COND STOR TK 1A LEVEL at Panel A.
   - PMIS Data Point F001.

   2.2.1 IF CST-A level after release cannot be determined, THEN **ASSUME** CST-A after release level of zero (0) feet.

   2.2.2 **RECORD** CST-A Level Following Release (ft) value in Table 12-1, Column 2.

2.3 **CALCULATE** CST-A volume released (ft) on Table 12-1 as follows:

2.3.1 For CST-A, **SUBTRACT** Column 2 from Column 1.

(Column 1) – (Column 2) = (Column 3)

2.3.2 **RECORD** CST-A Volume Released (ft) value in Table 12-1, Column 3.

2.4 **CALCULATE** CST-A volume released (Gallons) on Table 12-1 as follows:

2.4.1 **MULTIPLY** Column 3 by Column 4 to convert feet to gallons.

(Column 3) x (Column 4) = (Column 5)

2.4.2 **RECORD** CST-A Volume Released (gal) in Table 12-1, Column 5.

2.5 **PROCEED** to Step 5 to determine curie content released.

3. **USE** Table 12-1 to determine volume released from North Condensate Storage Tank (CST-B).

3.1 **OBTAIN** CST-B level (ft) prior to release from one of following:

- Control Room Indicator CM-LI-682, COND STOR TK 1B LEVEL at Panel A.
- PMIS Data Point N300.

3.1.1 IF CST-B level prior to release cannot be determined, THEN **ASSUME** CST-B was at maximum operational level of 34 feet.

3.1.2 **RECORD** CST-B Initial Level (ft) value in Table 12-1, Column 1.

3.2 **OBTAIN** CST-B level (ft) following release from one of following:

- Control Room Indicator CM-LI-682, COND STOR TK 1B LEVEL at Panel A.
- PMIS Data Point N300.

3.2.1 IF CST-B level after release cannot be determined, THEN **ASSUME** CST-B after release level of zero (0) feet.

3.2.2 **RECORD** CST-B Level Following Release (ft) value in Table 12-1, Column 2.

3.3 **CALCULATE** CST-B volume released (ft) on Table 12-1 as follows:

3.3.1 For CST-B, **SUBTRACT** Column 2 from Column 1.

(Column 1) – (Column 2) = (Column 3)

3.3.2 **RECORD** CST-B Volume Released (ft) value in Table 12-1, Column 3.

3.4 **CALCULATE** CST-B volume released (gallons) on Table 12-1 as follows:

3.4.1 **MULTIPLY** Column 3 by Column 4 to convert feet to gallons.

(Column 3) x (Column 4) = (Column 5)

3.4.2 **RECORD** CST-B Volume Released (gal) in Table 12-1, Column 5.

3.5 **PROCEED** to Step 5 to determine curie content released.

4. **USE** Table 12-2 to determine volume released from other tank(s) with potentially contaminated liquids.

4.1 **RECORD** selected tank(s) being analyzed on Table 12-2.

NOTE – Volume of tank before and after release may be obtained from Operations or by viewing PMIS data, if available.

4.2 **DETERMINE** volume of selected tank prior to release.

4.2.1 IF volume of tank prior to release cannot be determined, THEN **ASSUME** tank was 100%.

4.2.2 **RECORD** selected tank Initial Level (%) value in Table 12-2, Column 1.

4.3 **DETERMINE** volume of selected tank in percent after release was terminated.

4.3.1 IF volume of tank after release cannot be determined, THEN **ASSUME** tank was zero (0) %.

4.3.2 **RECORD** selected Tank Volume After Release (%) value in Table 12-2, Column 2.

4.4 **CALCULATE** selected tank volume released percent on Table 12-2 as follows:

4.4.1 **SUBTRACT** Column 2 from Column 1.

(Column 1) – (Column 2) = (Column 3)

4.4.2 **RECORD** selected Tank Volume Released (%) value in Table 12-2, Column 3.

4.5 **USE** Table 12-3 to determine capacity of selected tank.

4.5.1 **RECORD** selected Tank Capacity (gal) value in Table 12-3, Column 5.

4.6 **CALCULATE** selected tank volume released in gallons as follows:

4.6.1 **MULTIPLY** Column 3 by Column 4 by Column 5.

(Column 3) x (Column 4) x (Column 5) = (Column 6)

4.6.2 **RECORD** selected Tank Volume Released (gal) value in Table 12-3, Column 6.

4.7 **PROCEED** to Step 5 to determine curie content released.

5. **USE** Table 12-4 to determine gross activity of liquid being released from selected tank(s).

5.1 **RECORD** selected tank(s) being analyzed on Table 12-4.

5.2 **OBTAIN** Gross Activity Of Liquid (μCi/ml) value of selected tank(s) from Chem/RP Coordinator.

5.2.1 IF South Condensate Storage Tank (SCST) only and plant was operating normally prior to release,
THEN **USE** default value of 1.0E-2 μCi/ml.

5.2.2 **RECORD** Gross Activity Of Liquid (μCi/ml) value of selected tank(s) in Table 12-4, Column 1.

5.3 **MULTIPLY** Column 1 by Column 2 to convert selected tank(s) Gross Activity Of Liquid value to (μCi/gal).

(Column 1) x (Column 2) = (Column 3)

5.4 **RECORD** Gross Activity Of Liquid (μCi/gal) value of selected tank(s) in Table 12-4, Column 3.

6. **RECORD** Tank Volume Released of selected tank(s) in Table 12-4, Column 4.

6.1 IF CST-A or CST-B,
THEN **RECORD** Tank Volume Released (gal) value from Table 12-1, Column 5.

6.2 IF other selected tank,
THEN **RECORD** Tank Volume Released (gal) value from Table 12-2, Column 6.

7. **USE** Table 12-4 to determine Total Curies (Ci) released from selected tank(s).

7.1 **MULTIPLY** Column 3 by Column 4 to provide total microcuries (μCi) released from selected tank(s).

(Column 3) x (Column 4) = (Column 5)

7.1.1 **RECORD** Total μCi Released (μCi) value from selected tank(s) in Table 12-4, Column 5.

7.2 **MULTIPLY** Total μCi Released (μCi) value from selected tank(s) by Column 6 to convert to Curies (Ci).

7.2.1 **RECORD** Total Ci Released (Ci) value from selected tank(s) in Column 7.

7.2.2 IF more than one tank has release,
THEN **ADD** Column 7 Total Curies Released from each tank.

7.2.3 **RECORD** Total Curies Released in box at bottom of Table 12-4, Column 7.

8. **RECORD** Completed By and Date.

9. **INFORM** Chem/RP Coordinator in TSC.

10. **USE** Total Ci Released (Ci) value from Table 12-4, Column 7, to complete Off-Site Notification Form per EPIP 5.7.6.

11. **RECORD** results in facility log.

ATTACHMENT 12 LIQUID RELEASE CURIE CONTENT CALCULATION DATA SHEET (INFORMATION USE)

Table 12-1 - CONDENSATE STORAGE TANK LEVEL (FT) TO VOLUME RELEASED

CST	Maximum Volume (gal)	(1) Initial Level (ft)	(2) Level After Release (ft)	(3) Volume Released (ft) (1) - 2)	(4) Conversion Factor, ft to gal (gal/ft)	(5) CST Volume Released (gal) (3) x (4)
South Condensate Storage Tank (CST-A)	450,000				11,800	
North Condensate Storage Tank (CST-B)	700,000				19,090	

Table 12-2 - TANK LEVEL (%) TO VOLUME RELEASED

Tank	(1) Initial Tank Volume (%) (if not known, use 100%)	(2) Tank Volume After Release (%) (if not known, use 0%)	(3) Tank Volume Released (%) (1) - (2)	(4) Conv. Factor, % to decimal	(5) Selected Tank Capacity (gal) (see Table 12-3)	(6) Tank Volume Released (gal) (3) x (4) x (5)
				0.01		
				0.01		

Table 12-3 - CAPACITY OF TANKS (GAL) CONTAINING POTENTIALLY CONTAMINATED LIQUIDS

Tank	Capacity (gal)		Tank	Capacity (gal)
Floor Drain Sample Tank	20,000		Condensate Backwash Tank	12,500
Waste Sample Tank (each)	22,000		Condensate Phase Separators (each)	12,500
Waste Surge Tank	65,000		Waste Sludge Tank	17,000
Waste Collector Tank	22,000		RWCU Phase Separators (each)	4,500
Floor Drain Collector Tank	20,000		Spent Resin Tank	2,000
Chemical Waste Tank	5,200		Lab Drain Tank (each)	500
Torus/Suppression Pool	700,000			

Table 12-4 - LIQUID RELEASE CURIE CONTENT CALCULATION							
Tank	(1) Gross Activity of Liquid (μCi/ml)	(2) Conv. Factor (ml to gal)	(3) Gross Activity of Liquid (μCi/gal) (1) x (2)	(4) Tank Volume Released **ENTER** Table 12-1, Col. 5 <u>or</u> Table 12-2, Col. 6 (gallons)	(5) Total μCi Released (μCi) (3) x (4)	(6) Conv. Factor (μCi to Ci)	(7) Total Curies Released (Ci) per tank (5) x (6)
		3785				1E-6	
		3785				1E-6	
		3785				1E-6	
		3785				1E-6	
(8) Total Curies Released from all tanks =							

Completed By: ______________________________ Date: __________

ATTACHMENT 13 INFORMATION SHEET (INFORMATION USE)

1. PURPOSE ©[1]

1.1 Perform manual determination of airborne radioactive release rates from Elevated Release Point (ERP), Reactor Building, Turbine Building, and Radwaste Building vents, utilizing process monitor or other readings when CNS-DOSE program is not available.

1.2 Perform manual determination of total curie content from liquid releases.

2. PRECAUTIONS AND LIMITATIONS

2.1 Determination of ERP release rate using in-containment high range monitors should only be used if calculation cannot be performed utilizing ERP noble gas effluent monitor (KAMAN) readings.

2.2 When determining liquid release curie content and the selected tank initial or final tank levels cannot be determined, unless provided with an alternative value, assume total tank capacity was released.

3. DISCUSSION

3.1 KAMAN monitors are preferred means to determine airborne radioactive release rates and should be utilized if they can be restored to service.

3.2 Dose projections should be compared to field monitoring data, release point sample collection data, and other relevant data as it becomes available.

3.3 Upon determination of release rates, actual or projected plume exposure doses may be calculated per EPIPs 5.7.17 and 5.7.17.1.

3.3.1 These doses provide a basis for relating plume exposure doses to EPA Protective Action Guides (PAGs) per EPIP 5.7.20.

3.3.2 EPIP 5.7.17, CNS-DOSE Assessment, and EPIP 5.7.17.1, Dose Assessment (Manual), provide guidance for comparing off-site sample results with dose projections.

3.4 If ERP monitor becomes inoperable, ERP releases can be calculated using steam jet air ejector (SJAE) monitor readings if SJAEs are in release path.

3.5 If ERP monitor becomes inoperable, ERP releases can be calculated by correlating exposure rates on high range radiation monitors in Drywell to those which have been calculated assuming a Design Basis Accident Loss of Coolant Accident (DBA LOCA) with following assumptions:

3.5.1 DBA LOCA calculations are based on NUREG 0737 assumptions of: maximum full power equilibrium isotopic inventories, 100% of noble gases, 25% of halogens, and 1% of remaining radionuclides are instantaneously released to atmosphere of primary containment.

3.5.2 The leak rate from primary containment is assumed to be 0.105 volume/day (10.6 cfm).

3.5.3 Secondary containment purge rate is assumed to be 100% volume/day.

3.5.4 Entire release is assumed to be through SBGT System and out through ERP.

3.6 Noble gas, particulate, and iodine release rates may be verified or determined by sample collections from effluent release path and subsequent analysis in Radiochemistry Laboratory as directed by EPIP 5.7.19 under direction of Chemistry/Radiological Protection Coordinator.

3.7 In-containment activities, both liquid and gaseous, may be obtained to determine release concentration if release path is from containment as directed by EPIP 5.7.19 under direction of Chemistry/Radiological Protection Coordinator.

3.8 Other sampling procedures for determining release rates are also referenced in this procedure.

3.8.1 Activity levels used in Attachment 11, except for default value for South Condensate Storage Tank, will also be input to EFFECTS computer program to calculate downstream concentrations, dose rates, doses, and relationship to Off-Site Dose Assessment Manual release limits.

3.8.2 EFFECTS program is performed by Chemistry personnel.

3.8.3 Environmental Affairs group at Columbus General Office also has methodology programs for calculating exposures of liquid discharges.

4. RECORDS

4.1 Attachments below, for actual events, shall be forwarded to EP Manager within 5 working days of their completion (quality record upon completion):

- Attachment 3.
- Attachment 5.
- Attachment 7.
- Attachment 12.

5. REFERENCES

5.1 CODES AND STANDARDS

5.1.1 Environmental Protection Agency EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, May 1992.

5.1.2 NPPD Emergency Plan for CNS.

5.1.3 NUREG-0654/FEMA-REP-1, Revision 1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants.

5.1.4 NUREG-0737, Clarification of TMI Action Plan Requirements, November 1980.

5.2 PROCEDURES

5.2.1 Instrument Operating Procedure 4.15.1, Elevated Release Point Radiation Monitoring System.

5.2.2 Instrument Operating Procedure 4.15.2, Turbine Building Ventilation Radiation Monitoring System.

5.2.3 Instrument Operating Procedure 4.15.3, Radwaste Building Ventilation Radiation Monitoring System.

5.2.4 Instrument Operating Procedure 4.15.4, Reactor Building Ventilation Radiation Monitoring System.

5.2.5 Emergency Plan Implementing Procedure 5.7.6, Notification.

5.2.6 Emergency Plan Implementing Procedure 5.7.17.1, Dose Assessment (Manual).

5.2.7 Emergency Plan Implementing Procedure 5.7.19, On-Site Radiological Monitoring.

5.2.8 Emergency Plan Implementing Procedure 5.7.20, Protective Action Recommendations.

5.3 MISCELLANEOUS

5.3.1 NEDC 90-027, Flow Through 1" Vent Lines from Drywell and Torus at 65 psi Delta P.

5.3.2 NEDC 92-092, Review of Nutech Calculation of THPV Flow Rate and Vent Pipe Size.

5.3.3 NEDC 92-094, Review of Nutech Calculation of Hard Pipe Vent Pressure/Temperature Profile.

5.4 NRC Commitments

5.4.1 ©[1] NRC Commitment NLS2014035-03. Maintain I-131 site survey detection capability, including an ability to assess radioactive iodines released to off-site environs, by using effluent monitoring systems or portable sampling equipment. Commitment affects entire procedure (Attachment 13, Step 1 flagged).

Enclosure 4

Emergency Plan Implementing Procedure 5.7.17.1, Revision 0

COOPER NUCLEAR STATION

Operations Manual

Emergency Preparedness

EMERGENCY PLAN IMPLEMENTING PROCEDURE

5.7.17.1

DOSE ASSESSMENT (MANUAL)

Level of Use:	**MULTIPLE**
Quality:	**QAPD RELATED**
Effective Date:	**3/28/16**
Approval Authority:	**ITR-RDM**
Procedure Owner:	**EP ON-SITE COORDINATOR**

TABLE OF CONTENTS

1. ENTRY CONDITIONS (INFORMATION USE)

   1.1 Actual or projected radiological release above NOUE release rate limits has occurred requiring dose assessment per Emergency Director.

   1.2 CNS-DOSE software is unavailable or inoperable requiring use of hand calculated dose assessment.

2. INSTRUCTIONS (REFERENCE USE)

   2.1 **USE** Table 1 to select appropriate activity.

Table 1	
DESIRED ACTIVITY	GO TO
Dose assessment at plume centerline at 1, 2, 5, and 10 miles from site.	Attachment 1
Dose assessment at non-centerline specific receptor locations at non-standard distances from site.	Attachment 5
Core Damage Estimate using In-Containment High Range Radiation Monitors.©[1]	Attachment 9

   2.2 IF any of following occur,
   - Significant meteorological changes in wind speed, wind direction, and precipitation.
   - Every hour for effective age 0-10 hours.
   - Every 10 hours for effective age 10-100 hours.
   - Conditions reach critical parameter per Emergency Director (ED)/ Radiological Assessment Supervisor (RAS).

   THEN **RE-PERFORM** dose assessment.

ATTACHMENT 1 HAND CALCULATED DOSE PROJECTION (CENTERLINE) (REFERENCE USE)

NOTE 1 – Attachment 12 provides assistance for use of PMIS, including PMIS login and turn-on code activation.

NOTE 2 – Attachment 13 provides assistance for selection of PMIS RAD and MET data points.

1. IF CNS-DOSE unavailable but PMIS available, THEN **OBTAIN** RAD and MET data from following screens:
   - PMIS05.
   - SPDS01.
   - DOSEIN.
2. **OBTAIN** RAD release rates from following:
   - PMIS.
   - Effluent KAMAN monitor readouts in Control Room.
   - Effluent KAMAN monitor readouts in building.
   - Via calculation from EPIP 5.7.16.
3. IF KAMAN monitor inoperable, THEN concurrently **PERFORM** EPIP 5.7.16 to determine radiological release rate.
4. **RECORD** noble gas release rate value (μCi/sec) in Table 2-1, Block 1.

NOTE – SGT running and in flow path to ERP does not mean SGT is in effluent stream. Multiple release paths may exist (e.g., there is a release with SGT running in effluent stream to ERP with simultaneous unfiltered release through hard pipe vent). An unfiltered release pathway may necessitate using Step 5.4.©[2]

5. **DETERMINE** if release pathway through SGT.

   5.1 **CONSULT** with following to coordinate answer:©[2]
   - Operations.
   - RP/Chemistry, if available.
   - Engineering, if available.

5.2 Use Table 1-1 to **SELECT** appropriate entry for release pathway through SGT based on release path and plant conditions.

Table 1-1		
RELEASE SOURCE	RELEASE FILTERED THROUGH SGT	ENTER
ERP	Yes: Rx Bldg	0.01
ERP	No: Hard Pipe Vent	1.0
ERP	No: Gland Seal Steam, Main Condenser through Air Removal and Off-Gas	1.0
Reactor Bldg	No: Rx HVAC or direct ground level from blowout panel or other opening	1.0
Turbine Bldg	No: TB HVAC or direct ground level from blowout panel or other opening	1.0
Turbine Bldg	No: TB HVAC plus Reactor Bldg via TB-Reactor Bldg blowout panel	1.0
RW/ARW Bldg	No: RW/ARW HVAC or other direct ground level opening	1.0

5.3 IF release pathway through SGT,
THEN **RECORD** 0.01 in Table 2-1, Block 2.

5.4 IF release pathway not through SGT,
THEN **RECORD** 1.0 in Table 2-1, Block 2.

NOTE – Reactor shutdown under all conditions is met by any of following conditions:
- All rods fully inserted.
- Reactor scram and power below 3%.
- Cold shutdown boron weight injected per EOPs.

6. **OBTAIN** time of Rx Shutdown under all conditions.

   6.1 **RECORD** time of Rx Shutdown: ________.

   6.2 **CONSULT** with following to coordinate answer:
   - Operations.
   - RP/Chemistry, if available.
   - Engineering, if available.

   6.3 **RECORD** Time Since Shutdown under all conditions in hours value in Table 2-1, Block 3.

7. **DETERMINE** if core is degraded based on plant conditions.

   7.1 **REFER** to Attachment 8 for guidance on degraded core.©[4]

   7.2 **CONSULT** with following to coordinate answer:©[5]
   - Operations.
   - RP/Chemistry, if available.
   - Engineering, if available.

   7.3 IF core degraded,
   THEN **OBTAIN** Iodine to Noble Gas ratio from Table 2-7.

   7.3.1 **RECORD** value in Table 2-1, Block 4.

   7.4 IF core not degraded,
   THEN **RECORD** 1.86E-7 in Table 2-1, Block 4.

8. **OBTAIN** Noble Gas Energy Factor (MeV/dis) based on time since shutdown in hours from Table 2-8.

   8.1 **RECORD** value in Table 2-1, Block 5.

NOTE – MET Tower instrumentation from different trains may be used for each elevation or parameter when performing dose assessment with "A" being preferred train. If 100 meter MET tower is out of service or unavailable, PMIS will default to next highest tower if elevated release. Default is 60M.

9. **OBTAIN** 15 minute average wind speed in miles per hour (mph).

 9.1 IF release from ERP,
 THEN **USE** data in following preferred order for elevated release:

  9.1.1 PMIS 100 meter level.

  9.1.2 PMIS 60 meter level.

  9.1.3 **CONTACT** NWS at 402-359-4381 or refer to telephone directory and **OBTAIN** average wind speed estimate for 100 meter elevation.

  9.1.4 Historical default wind speed value of 13 mph.

 9.2 IF release from any other source,
 THEN **USE** data in following preferred order for ground level release:

  9.2.1 PMIS 10 meter level.

  9.2.2 PMIS 60 meter level.

  9.2.3 **CONTACT** NWS at 402-359-4381 or refer to telephone directory and **OBTAIN** average wind speed estimate for 10 meter elevation.

  9.2.4 Historical default wind speed value of 8 mph.

 9.3 **RECORD** 15 minute average wind speed in Table 2-1, Block 6.

10. **OBTAIN** wind direction (degree from).

10.1 IF release source from ERP,
THEN **USE** data in following preferred order:

10.1.1 PMIS 100 meter level.

10.1.2 PMIS 60 meter level.

10.1.3 **CONTACT** NWS at 402-359-4381 or refer to telephone directory and **OBTAIN** estimate of average wind direction at CNS (degree from) at 100 meter elevation.

10.2 IF release source from any other source,
THEN **USE** data in following preferred order:

10.2.1 PMIS 10 meter level.

10.2.2 PMIS 60 meter level.

10.2.3 **CONTACT** NWS at 402-359-4381 or refer to telephone directory and **OBTAIN** estimate of average wind direction at CNS (degree from) for 10 meter elevation.

10.3 **RECORD** wind direction (degree from) in Table 2-1, Block 7.

11. **DETERMINE** atmospheric stability class ("A" through "G").

11.1 IF release from ERP,
THEN **USE** data in following preferred order:

11.1.1 Direct PMIS stability class readout.

11.1.2 PMIS data below and Table 1-2.

100 meter (°F) _____ minus 10 meter (°F) ______ equals delta-T ______

Table 1-2

Delta-T °F	≤ -3.08	> -3.08 to ≤ -2.75	> -2.75 to ≤ -2.43	> -2.43 to ≤ -0.81	> -0.81 to ≤ 2.43	> 2.43 to ≤ 6.48	> 6.48
Stability Class	A	B	C	D	E	F	G

11.1.3 PMIS data below and Table 1-3.

60 meter (°F) ______ minus 10 meter (°F) ______ equals delta-T ______

Table 1-3							
Delta-T °F	≤ -1.71	> -1.71 to ≤ -1.46	> -1.46 to ≤ -1.35	> -1.35 to ≤ -0.45	> -0.45 to ≤ 1.35	> 1.35 to ≤ 3.6	> 3.6
Stability Class	A	B	C	D	E	F	G

11.1.4 PMIS 100 meter sigma-theta data below and Table 1-4.

11.1.5 PMIS 60 meter sigma-theta data below and Table 1-4.

11.1.6 PMIS 10 meter sigma-theta data below and Table 1-4.

Sigma-theta from 100 M, 60 M, or 10 M: ______________

Table 1-4							
Sigma-Theta	≥ 22.5	< 22.5 to ≥ 17.5	< 17.5 to ≥ 12.5	< 12.5 to ≥ 7.5	< 7.5 to ≥ 3.8	< 3.8 to ≥ 2.1	< 2.1
Stability Class	A	B	C	D	E	F	G

11.1.7 NWS data below and Table 1-5.

11.1.7.1 **CONTACT** NWS at 402-359-4381 or refer to telephone directory.

100 meter (°F) ______ minus 10 meter (°F) ______ equals delta-T ______

Table 1-5							
Delta-T °F	≤ -3.08	> -3.08 to ≤ -2.75	> -2.75 to ≤ -2.43	> -2.43 to ≤ -0.81	> -0.81 to ≤ 2.43	> 2.43 to ≤ 6.48	> 6.48
Stability Class	A	B	C	D	E	F	G

11.1.8 <u>IF</u> stability class unknown,
<u>THEN</u> **USE** historical default "D" stability class.

11.2 IF release from any other source,
THEN **USE** data in following preferred order:

11.2.1 Direct PMIS stability class readout.

11.2.2 PMIS data below and Table 1-6.

60 meter (°F) ______ minus 10 meter (°F) ______ equals delta-T ______

Table 1-6							
Delta-T °F	≤ -1.71	> -1.71 to ≤ -1.46	> -1.46 to ≤ -1.35	> -1.35 to ≤ -0.45	> -0.45 to ≤ 1.35	> 1.35 to ≤ 3.6	> 3.6
Stability Class	A	B	C	D	E	F	G

11.2.3 PMIS data below and Table 1-7.

100 meter (°F) ______ minus 10 meter (°F) ______ equals delta-T ______

Table 1-7							
Delta-T °F	≤ -3.08	> -3.08 to ≤ -2.75	> -2.75 to ≤ -2.43	> -2.43 to ≤ -0.81	> -0.81 to ≤ 2.43	> 2.43 to ≤ 6.48	> 6.48
Stability Class	A	B	C	D	E	F	G

11.2.4 PMIS 10 meter sigma-theta data below and Table 1-8.

11.2.5 PMIS 60 meter sigma-theta data below and Table 1-8.

11.2.6 PMIS 100 meter sigma-theta data below and Table 1-8.

Sigma-theta from 10 M, 60 M, or 100 M: ______________

Table 1-8							
Sigma-Theta	≥ 22.5	< 22.5 to ≥ 17.5	< 17.5 to ≥ 12.5	< 12.5 to ≥ 7.5	< 7.5 to ≥ 3.8	< 3.8 to ≥ 2.1	< 2.1
Stability Class	A	B	C	D	E	F	G

11.2.7 NWS data below and Table 1-9.

11.2.7.1 **CONTACT** NWS at 402-359-4381 or refer to telephone directory.

60 meter (°F) _______ minus 10 meter (°F) ______ equals delta-T ______

Table 1-9							
Delta-T °F	≤ -1.71	> -1.71 to ≤ -1.46	> -1.46 to ≤ -1.35	> -1.35 to ≤ -0.45	> -0.45 to ≤ 1.35	> 1.35 to ≤ 3.6	> 3.6
Stability Class	A	B	C	D	E	F	G

11.2.8 IF stability class unknown,
THEN **USE** historical default "D" stability class.

11.3 **RECORD** stability class value in Table 2-1, Block 8.

12. **DETERMINE** if release pathway is into Reactor Building (credit Rx BLDG for plate-out).

12.1 **CONSULT** with following to coordinate answer:

- Operations.
- RP/Chemistry, if available.
- Engineering, if available.

12.2 Use Table 1-10 to **SELECT** appropriate entry for release pathway based on release path and plant conditions.

Table 1-10		
RELEASE SOURCE	RELEASE PATH	ENTER
ERP	Reactor Bldg through SGT without Hard Pipe Vent	Yes-0.5
ERP	Primary Containment through Hard Pipe Vent	No-1.0
ERP	Gland Seal Steam, Main Condenser through Air Removal, and Off-Gas	No-1.0
Reactor Bldg	Reactor Bldg HVAC or direct ground level from blowout panel or other opening	Yes-0.5
Turbine Bldg	TB HVAC or direct ground level from blowout panel or other opening	No-1.0
Turbine Bldg	TB HVAC plus Reactor Bldg via TB-Reactor Bldg blowout panel	Yes-0.5
RW/ARW Bldg	RW/ARW HVAC or other direct ground level opening	No-1.0

12.3 IF NO indicating release bypasses Reactor Building, THEN **RECORD** 1.0 in Table 2-1, Block 9.

12.4 IF YES indicating release path into or through Reactor Building, THEN **RECORD** 0.5 in Table 2-1, Block 9.

13. **OBTAIN** TEDE Noble Gas Dose Conversion Factor from Table 2-9.

    13.1 **RECORD** value in Table 2-2, Block 10.

14. **OBTAIN** TEDE Iodine Dose Conversion Factor from Table 2-9.

    14.1 **RECORD** value in Table 2-2, Block 11.

15. **OBTAIN** CDE Iodine Dose Conversion Factor from Table 2-9.

    15.1 **RECORD** value in Table 2-2, Block 12.

NOTE – Control Room personnel are not responsible for correlating Field Monitoring Team dose rate readings with hand calculated dose projections.

16. **CALCULATE** TEDE "Sub-Calculation" value.

16.1 **MULTIPLY** Table 2-1, Block 1 by Block 5, and then by Table 2-2, Block 10, to obtain Noble Gas portion.

16.2 **MULTIPLY** Table 2-1, Block 1 by Block 2 by Block 4 by Block 9, and then by Table 2-2, Block 11, to obtain Iodine portion.

16.3 **ADD** Noble Gas results from Step 16.1 and Iodine results from Step 16.2.

16.4 **DIVIDE** results of Step 16.3 by Table 2-1, Block 6, to complete TEDE Sub-Calculation.

NOTE – Hand calculated dose assessment projections shall not be corrected downward based on field samples.

16.5 **OBTAIN** Field Sample Correction Factor (CF) results from Table 4-1, Column 6.

16.5.1 IF Field Sample CF results unavailable or less than 1.0, THEN **RECORD** 1.0 and **PROCEED** to Step 16.6.

16.5.2 IF Field Sample CF results greater than 1.0, THEN **PERFORM** following:

16.5.2.1 **RECORD** Field Sample CF value in Table 2-3, upper Block 13.

16.5.2.2 **MULTIPLY** results of Step 16.4 by Field Sample CF to raise TEDE dose rate assessment.

16.6 **RECORD** TEDE Sub-Calculation value in Table 2-3, Block 13.

17. **OBTAIN** mixing factors (χ/Qs) for distances 1, 2, 5, and 10 miles from Table 2-10.©[10]

    17.1 **USE** appropriate release point (ERP or other than ERP).

    17.2 **USE** stability class from Table 2-1, Block 8.

    17.3 **RECORD** mixing factors (χ/Qs) values in Table 2-2, Block 14.

18. **CALCULATE** TEDE dose rate (rem/hr) for each distance 1, 2, 5, and 10 miles.

    18.1 **MULTIPLY** Table 2-3, Block 13, by Table 2-2, Block 14.

    18.2 **RECORD** TEDE dose rate (rem/hr) values in Table 2-3, Block 15.

19. **ESTIMATE** Duration of Release in hours.

    19.1 **CONSULT** with following to coordinate answer:

    - Operations.
    - RP/Chemistry, if available.
    - Engineering, if available.

    19.2 IF duration unknown,
    THEN **USE** 4 hour default value.

    19.3 **RECORD** estimated duration hours in Table 2-3, Block 16.

20. **CALCULATE** integrated TEDE dose (rem) for each distance 1, 2, 5, and 10 miles.

    20.1 **MULTIPLY** Table 2-3, Block 15 by Block 16.

    20.2 **RECORD** TEDE dose (rem) values in Table 2-3, Block 17.

    20.3 IF higher level emergency classification is discovered after calculating TEDE dose values in Block 17,
    THEN immediately **INFORM** ED or RCM.

21. **CALCULATE** CDE "Sub-Calculation" value.

    21.1 **MULTIPLY** Table 2-1, Block 1 by Block 2 by Block 4 by Block 9.

    21.2 **MULTIPLY** results of Step 21.1 by Table 2-2, Block 12.

    21.3 **DIVIDE** results by Table 2-1, Block 6.

NOTE – Hand calculated dose assessment projections shall not be corrected downward based on field samples.

    21.4 **OBTAIN** Field Sample Correction Factor (CF) results from Table 4-1, Column 6.

        21.4.1 IF Field Sample CF results unavailable or less than 1.0, THEN **RECORD** 1.0 and **PROCEED** to Step 21.5.

        21.4.2 IF Field Sample CF results greater than 1.0, THEN **PERFORM** following:

            21.4.2.1 **RECORD** Field Sample CF value into upper Table 2-4, Block 18.

            21.4.2.2 **MULTIPLY** results of Step 21.3 by Field Sample CF to raise CDE dose rate assessment.

    21.5 **RECORD** CDE Sub-Calculation value in Table 2-4, Block 18.

22. **CALCULATE** CDE dose rate (rem/hr) for each distance 1, 2, 5, and 10 miles.

    22.1 **MULTIPLY** Table 2-2, Block 14, by Table 2-4, Block 18.

    22.2 **RECORD** CDE dose rate (rem/hr) values in Table 2-5, Block 19.

23. **CALCULATE** CDE dose (rem) for each distance 1, 2, 5, and 10 miles.

    23.1 **MULTIPLY** Table 2-3, Block 16, by Table 2-5, Block 19.

    23.2 **RECORD** CDE dose (rem) values in Table 2-6, Block 20.

24. **RECORD** name, time, and date at bottom of Attachment 2.

    24.1 IF Attachment 2 was completed electronically, THEN **PRINT** Attachment 2.

25. IF multiple release points are identified, THEN **DETERMINE** if summation of various releases could result in PAR change or change in emergency classification level.

    25.1 **PERFORM** dose assessment for each release point by repeating Steps 2 through 24.

    25.2 **OBTAIN** copy of Excel spreadsheet from EP webpage under Dose Assessment tab using business computer.

    25.3 **SAVE** spreadsheet to user directory of business computer.

    25.4 **DEPRESS** Clear Input Screen button on Input tab.

    25.5 **RECORD** following from completed Attachment 2 on Input tab in proper column based on release point:

        25.5.1 Release Rate from Table 2-1, Block 1.

        25.5.2 Integrated TEDE dose from Table 2-3, Block 17, and Integrated CEDE dose from Table 2-6, Block 20.

        25.5.3 TEDE dose rate from Table 2-3, Block 15, and CED dose rate from Table 2-5, Block 19.

    25.6 **RECORD** wind directions for Elevated Release Point and Ground Level, as appropriate, for multiple releases.

    25.7 WHEN all identified release points have been entered, THEN **DEPRESS** Print Package button.

        25.7.1 **ENSURE** all pages are printed.

        25.7.2 **RECORD** times on appropriate pages.

    25.8 IF spreadsheet cannot be obtained, saved, or used, THEN **PERFORM** manual summation of dose assessments based on each sector and distance utilizing Attachment 10 or similar data sheet.

26. IF in Control Room, THEN **NOTIFY** SM/ED dose projection(s) and multi-source releases, if required, are complete.

27. IF in EOF,
THEN **NOTIFY** Radiological Assessment Supervisor (RAS) and RCM dose projection(s) and multi-source releases, if required, are complete.

27.1 RAS or RCM shall **REVIEW** completed dose projection(s).

27.2 RCM **DETERMINE** if emergency classification or PAR is affected by radiological effluent (dose rate or integrated dose to a member of public) calculated at or beyond 1 mile.

27.2.1 RCM **REFER** to Technical Bases Effluent Monitor Classification Thresholds Table in EPIP 5.7.1.

27.2.2 RCM **RECOMMEND** emergency classification or PAR change to ED.

NOTE – Control Room personnel are not responsible for correlating Field Monitoring Team dose rate readings with Hand Calculated dose projections.

28. WHEN Field Monitoring Team reports air sample results at receptor location,
THEN **PERFORM** following to Attachments 3 and 4:©[3]

28.1 **COMPARE** field sample readings to hand calculated centerline dose projection results of Attachment 2.

28.2 **APPLY** Field Sample Correction Factor, as applicable, to adjust hand calculated centerline dose calculation accordingly.

29. **NOTIFY** RCM field samples comparison to hand calculated dose projections is complete.

30. RCM **RECOMMEND** if change to PARs necessary.

ATTACHMENT 2 HAND CALCULATED DOSE PROJECTION (CENTERLINE) WORKSHEET (INFORMATION USE)

Table 2-1

(1) Noble Gas Release Rate (μCi/Sec)	(2) Release Path through SGT?	(3) Time Since Shutdown	(4) Iodine/Noble Gas Ratio (from Table 2-7)	(5) Noble Gas Energy Factor (MeV/dis) (from Table 2-8)	(6) Wind Speed (mph)	(7) Wind Direction (° from)	(8) Stability Class	(9) Release through Reactor Building?

Table 2-2

Conversion Factors (from Table 2-9)	
TEDE Noble Gas (10)	
TEDE Iodine (11)	
CDE Iodine (12)	
Mixing Factors (from Table 2-10)©[10]	
1 Mile (14)	
2 Mile (14)	
5 Mile (14)	
10 Mile (14)	

Table 2-3

(13) TEDE Sub-Calculation

$$\frac{[(\text{Block 1})(\text{Block 5})(\text{Block 10})]+[(\text{Block 1})(\text{Block 2})(\text{Block 4})(\text{Block 9})(\text{Block 11})]}{(\text{Block 6})}$$ x Field sample CF: (Table 4-1)

(13)

TEDE Rate (rem/hr) (Block 13 x Block 14)		Duration (hours) (16)	TEDE Dose (rem) (Block 15 x Block 16)	
1 Mile (15)			1 Mile (17)	
2 Mile (15)			2 Mile (17)	
5 Mile (15)			5 Mile (17)	
10 Mile (15)			10 Mile (17)	

Table 2-4

(18) CDE Sub-Calculation

$$\frac{[(\text{Block 1})(\text{Block 2})(\text{Block 4})(\text{Block 9})(\text{Block 12})]}{(\text{Block 6})}$$ x Field sample CF: (Table 4-1)

(18)

Table 2-5

CDE Rate (rem/hr) (Block 14 x Block 18)	
1 Mile (19)	
2 Mile (19)	
5 Mile (19)	
10 Mile (19)	

Table 2-6

CDE Dose (rem) (Block 16 x Block 19)	
1 Mile (20)	
2 Mile (20)	
5 Mile (20)	
10 Mile (20)	

Name/Time/Date: ______________ / ______________ / ______________

Table 2-7: Iodine To Noble Gas Ratio vs. Time Since Shutdown

TIME SINCE SHUTDOWN (hrs)	IODINE/NOBLE GAS RATIO	
	NON-DEGRADED CORE	DEGRADED CORE
t < 1	1.86 E-7	2.71 E-1
1 ≤ t < 2		3.57 E-1
2 ≤ t < 4		3.41 E-1
4 ≤ t < 10		2.81 E-1
10 ≤ t < 30		2.30 E-1
30 ≤ t < 100		1.65 E-1
100 ≤ t		1.40 E-1

Table 2-8: Noble Gas Energy Factors

TIME SINCE SHUTDOWN (hrs)	ENERGY FACTOR (MeV/dis)
t < 1	0.75
1 ≤ t < 2	0.60
2 ≤ t < 4	0.40
4 ≤ t < 10	0.25
10 ≤ t < 30	0.15
30 ≤ t < 100	0.09
100 ≤ t	0.07

Table 2-9: Dose Conversion Factors

	NON-DEGRADED CORE	DEGRADED CORE
TEDE Noble Gas	1.48 E-3	9.19 E-4
TEDE Iodine	8.77 E-2	2.98 E-2
CDE Iodine	2.04 E 0	4.96 E-1

Table 2-10: Plume Centerline χ/Q's (Mixing Factors)

RELEASE POINT	STABILITY CLASS	A	B	C	D	E	F	G
ERP (ELEVATED)	1 MILE	2.87E-6	6.04E-6	1.17E-5	8.35E-6	1.03E-6	2.35E-11	1.31E-23
	2 MILE	7.94E-7	1.78E-6	4.55E-6	8.21E-6	4.98E-6	8.12E-8	5.62E-13
	5 MILE	1.50E-7	3.42E-7	1.18E-6	3.77E-6	4.66E-6	1.09E-6	5.67E-9
	10 MILE	4.51E-8	1.03E-7	4.58E-7	1.82E-6	3.13E-6	1.44E-6	4.00E-8
OTHER THAN ERP (GROUND LEVEL)	1 MILE	3.01E-6	6.90E-6	1.73E-5	5.10E-5	1.09E-4	3.07E-4	7.67E-4
	2 MILE	8.03E-7	1.84E-6	5.15E-6	1.78E-5	3.86E-5	1.09E-4	2.71E-4
	5 MILE	1.50E-7	3.44E-7	1.21E-6	4.98E-6	1.25E-5	3.52E-5	8.81E-5
	10 MILE	4.51E-8	1.03E-7	4.63E-7	2.07E-6	6.43E-6	1.81E-5	4.52E-5

ATTACHMENT 3 CORRELATING OFF-SITE SAMPLE RESULTS WITH HAND CALCULATED DOSE PROJECTIONS (CENTERLINE) (REFERENCE USE)©[3]

NOTE 1 – Initial hand calculated dose projections are based on assumed radionuclide concentrations until actual concentrations have been measured in field.

NOTE 2 – Field sample results are used to determine a dose rate correction factor which may be applied to adjust PARs.

1. **PERFORM** adjustments to Field Monitoring Team air sample results to correspond to plume centerline, receptor location if non-centerline dose projection, or distance of 1, 2, 5, or 10 miles, as follows:

   1.1 IF Field Monitoring Team samples are aligned to centerline of wind direction or were taken at highest radiological point during traverse of plume,
   THEN **GO TO** Step 1.3.

   1.2 **PERFORM** following to determine non-centerline adjustment ratio for field sample:

   1.2.1 **RECORD** field sample dose rate reading in Table 3-1, Column (1).

   1.2.2 **SELECT** isopleth based on atmospheric stability class.

   NOTE – Isopleth overlays are designed to scale to only 10 mile EPZ map with 1 mile to 1 inch scale. Additional extrapolation required if applying to 50 mile EPZ map.

   1.2.3 **PLACE** selected isopleth onto 10 mile EPZ map with isopleth asterisk zero directly over zero point on map representing plant.

   1.2.3.1 **VERIFY** map and isopleth align to correct scale.

   1.2.4 **OBTAIN** downwind direction during period associated with field sample.

   1.2.5 **ALIGN** isopleth centerline on 10 mile EPZ map according to downwind direction centerline.

   1.2.6 **LOCATE** field sample location on 10 mile EPZ map.

1.2.7 **RECORD** alphabetic designation of isopleth segment of plume in Column (2) of Table 3-1 that field sample is located in.

NOTE – Interpolation between isopleth segment χ/Q values may be used to increase accuracy when sample location lies between isopleth segment lines.

1.2.8 **USE** χ/Q VALUES table, ISOPLETHS column, to determine χ/Q value associated with field sample isopleth segment recorded in Table 3-1, Column (2).

1.2.8.1 **RECORD** χ/Q value associated with isopleth segment containing field sample location in Table 3-1, Column (3).

1.2.9 At same distance as field sample location from release point, **DETERMINE** alphabetic designation of isopleth segment of plume centerline.

1.2.9.1 **RECORD** alphabetic designation of isopleth segment of plume centerline in Table 3-1, Column (4).

1.2.10 **USE** χ/Q VALUES table, ISOPLETHS column, to determine χ/Q value associated with isopleth segment of plume centerline recorded in Table 3-1, Column (4).

1.2.10.1 **RECORD** χ/Q value associated with isopleth segment of plume centerline in Table 3-1, Column (5).

NOTE – χ/Q value associated with each isopleth segment away from centerline is factor of 10 lower than centerline. Therefore, to adjust off-centerline sample to centerline, sample must be multiplied by factors of 10 based on number of segments away from centerline.

1.2.11 **DETERMINE** plume centerline adjustment ratio between field sample χ/Q value and plume centerline χ/Q value.

1.2.11.1 **DIVIDE** χ/Q value associated with isopleth segment of plume centerline in Table 3-1, Column (5), by χ/Q value associated with isopleth segment containing field sample location in Column (3).

1.2.11.2 **RECORD** plume centerline adjustment ratio in Table 3-1, Column 6.

1.2.12 **MULTIPLY** field sample reading recorded in Table 3-1, Column (1), by plume centerline adjustment ratio Column (6) to determine non-centerline adjusted field sample.

1.2.12.1 **RECORD** non-centerline adjusted field sample results in Table 3-1, Column 7.

Table 3-1

(1) Field Sample Dose Rate (rem/hr)	(2) Isopleth Segment at Sample Location (A-Q)	(3) χ/Q Value for Isopleth Sector at Sample Location	(4) Isopleth Segment at Plume Centerline at Sample Distance (A-Q)	(5) χ/Q Value for Isopleth Sector at Plume Centerline	(6) Plume Centerline Adjustment Ratio (5) ÷ (3) (>1.0)	(7) Non-Centerline Adjusted Field Sample Dose Rate (1) × (6)

1.2.13 <u>IF</u> sample distance associated with adjusted non-centerline field sample results in Table 3-1, Column (7), is aligned to distances of 1, 2, 5, <u>or</u> 10 miles,
<u>THEN</u> **PERFORM** Attachment 4 using adjusted field sample dose rate results in Table 3-1, Column (7), to compare against hand calculated dose rate at same distance.

1.2.14 <u>IF</u> sample distance associated with adjusted non-centerline field sample results in Table 3-1, Column (7), is <u>not</u> aligned to distances of 1, 2, 5, <u>or</u> 10 miles,
<u>THEN</u> **PERFORM** Step 1.3 to adjust non-centerline field sample dose rate to hand calculated distances.

NOTE – Adjusting centerline to 1 or 2 miles from 1.5 mile sample point centerline will not change dose assessment results, either is acceptable.

1.3 **PERFORM** distance adjustment of either field sample or non-centerline adjusted field sample dose rate results recorded in Column (7) of Table 3-1.

1.3.1 **RECORD** one of following in Table 3-2, Column (1):

- Direct field sample concentration whose location was on plume centerline.
- Adjusted non-centerline field sample results from Table 3-1, Column (7).

1.3.2 **RECORD** actual field sample distance from release source in miles in Table 3-2, Column (2).

1.3.3 **USE** χ/Q VALUES table, CENTERLINE column, to determine χ/Q value associated with actual field sample distance from release source in miles recorded in Table 3-2, Column (2).

1.3.3.1 **RECORD** χ/Q value in Table 3-2, Column (3), associated with actual field sample distance recorded in Column (2).

1.3.4 **RECORD** standard distance of 1, 2, 5, or 10 miles associated with hand calculated projection results being evaluated in Table 3-2, Column (4).

1.3.5 At χ/Q VALUES table, **USE** CENTERLINE column to determine χ/Q value associated with distance recorded in Table 3-2, Column (4).

1.3.5.1 **RECORD** χ/Q value in Table 3-2, Column (5), associated with hand calculated distance recorded in Column (4).

1.3.6 **DIVIDE** χ/Q value associated with distance in Table 3-2, Column (5), by χ/Q value associated with actual field sample distance in Column (3) to determine centerline distance adjustment ratio.

1.3.6.1 **RECORD** Centerline Distance Adjustment Ratio in Table 3-2, Column 6.

1.3.7 **MULTIPLY** field sample or adjusted field sample recorded in Table 3-2, Column 1, by Centerline Distance Adjustment Ratio in Column 6.

1.3.7.1 **RECORD** Adjusted Field Sample dose rate reading in Table 3-2, Column 7.

Table 3-2

(1) Field Sample or Non-Centerline Adjusted Field Sample [Table 3-1, Column (7)] (rem/hr)	(2) Actual Sample Distance on Plume Centerline (1 - 10 miles)	(3) Centerline Column χ/Q Value Associated with Actual Field Sample Distance	(4) Hand Calculated Distance (1, 2, 5, or 10)	(5) Centerline Column χ/Q Value Associated with Hand Calculated Distance	(6) Centerline Distance Adjustment Ratio (5) ÷ (3)	(7) Adjusted Field Sample (Dose Rate) (1) × (6)

1.4 **PERFORM** Attachment 4 using Adjusted Field Sample dose rate in Table 3-2, Column 7, to compare against hand calculated dose rate projection at same distance.

ATTACHMENT 4 CORRECTION DETERMINATION USING OFF-SITE SAMPLE RESULTS WITH DOSE PROJECTIONS (CENTERLINE) (REFERENCE USE)©3

1. **COMPLETE** Field Monitoring Team sample information in Table 4-1.
   - **CIRCLE** appropriate field sample downwind distance in Column (1).
   - **RECORD** field sample downwind segment (A-Q) in Column (2).
   - **RECORD** field sample time in military format (0000 hrs) in Column (3).
   - **RECORD** TEDE dose Rate (rem/hr) in Column (4) from direct field sample or correlated field sample recorded in Table 3-2, Column 7.
2. In Table 4-1, Column (5), **RECORD** hand calculated (Centerline) projected TEDE dose rate (rem/hr) from Table 2-3, Block 15, for same downwind distance as field sample location selected in Column (1).
3. **DIVIDE** Column (4) by Column (5) and **RECORD** Field Sample Correction Factor (CF) in Table 4-1, Column (6).
4. **RECORD** Initials, Date, and Time of comparison.
5. **PROVIDE** Attachment 4 to RAS or RCM for review.
6. IF Field Sample CF in Table 4-1, Column (6), is less than 1.0, THEN **GO TO** Step 8.
7. IF Field Sample CF in Table 4-1, Column (6), is greater than 1.0, indicating field sample result is greater than hand calculated dose rate projection, THEN **PERFORM** following:

   7.1 RAS or RCM **APPROVE** Field Sample CF results in Table 4-1, Column (6), to adjust hand calculated dose assessment.

   7.1.1 **APPLY** approved Field Sample CF during subsequent performance of Attachment 1.

8. RCM **DETERMINE** if emergency classification or PAR is affected by radiological effluent (dose rate or integrated dose to a member of public) calculated at or beyond 1 mile.

   8.1 RCM **REFER** to Technical Bases Effluent Monitor Classification Thresholds Table in EPIP 5.7.1.

   8.2 RCM **RECOMMEND** emergency classification or PAR change to Emergency Director.

Table 4-1

(1) Field Sample Location Miles Downwind (Circle One)	(2) Field Sample Location Downwind Segment (A-Q)	(3) Field Sample Time (0000 hrs)	(4) Correlated or Direct Field Sample Dose Rate Reading (rem/hr)	(5) Hand Calculated (Centerline) Dose Rate (rem/hr)	(6) Field Sample Correction Factor (CF)* (Column 4 ÷ Column 5)	Initials/Date/Time
1 2 5 10						____/____/____
1 2 5 10						____/____/____
1 2 5 10						____/____/____
1 2 5 10						____/____/____

* Field Sample Correction Factor (CF) results greater than 1.0 indicates hand calculation results are non-conservative. This condition is resolved by application of Field Sample Correction Factor during subsequent hand calculated dose projections.

ATTACHMENT 5 HAND CALCULATED DOSE PROJECTION (NON-CENTERLINE) FROM EOF (REFERENCE USE)

NOTE – Attachment 12 provides assistance for use of PMIS, including PMIS login and turn-on code activation.

1. IF PMIS available,
   THEN **OBTAIN** RAD and MET data from following screens:
   - PMIS05.
   - SPDS01.
   - DOSEIN.
2. **OBTAIN** noble gas release rates from following:
   - PMIS
   - Effluent KAMAN monitor readouts in Control Room.
   - Effluent KAMAN monitor readouts in building.
   - Via calculation from EPIP 5.7.16.
3. IF KAMAN monitor inoperable,
   THEN concurrently **PERFORM** EPIP 5.7.16 to determine radiological release rate.
4. **RECORD** noble gas release rate value (μCi/sec) in Table 6-1, Block 1.

NOTE – SGT running and in flow path to ERP does not mean SGT is in effluent stream. Multiple release paths may exist (e.g., there is a release with SGT running in effluent stream to ERP with simultaneous unfiltered release through hard pipe vent). An unfiltered release pathway may necessitate using Step 5.4.©2

5. **DETERMINE** release pathway through SGT.

   5.1 **CONSULT** with following to coordinate answer:©2
   - Operations.
   - RP/Chemistry, if available.
   - Engineering, if available.

5.2 Use Table 5-1 to **SELECT** appropriate entry for release pathway through SGT based on release path and plant conditions.

Table 5-1		
RELEASE SOURCE	RELEASE FILTERED THROUGH SGT	ENTER
ERP	Yes: Rx Bldg	0.01
ERP	No: Hard Pipe Vent	1.0
ERP	No: Gland Seal Steam, Main Condenser through Air Removal, and Off-Gas	1.0
Reactor Bldg	No: Rx HVAC or direct ground level from blowout panel or other opening	1.0
Turbine Bldg	No: TB HVAC or direct ground level from blowout panel or other opening	1.0
Turbine Bldg	No: TB HVAC plus Reactor Bldg via TB-Reactor Bldg blowout panel	1.0
RW/ARW Bldg	No: RW/ARW HVAC or other direct ground level opening	1.0

5.3 IF release pathway through SGT,
THEN **RECORD** 0.01 in Table 6-1, Block 2.

5.4 IF release pathway not through SGT,
THEN **RECORD** 1.0 in Table 6-1, Block 2.

NOTE – Reactor shutdown under all conditions is met by any of following conditions:
- All rods fully inserted.
- Reactor scram and power below 3%.
- Cold shutdown boron weight injected per EOPs.

6. **OBTAIN** time of Rx Shutdown under all conditions.

   6.1 **RECORD** time of Rx Shutdown: _______.

   6.2 **CONSULT** with following to coordinate answer:
   - Operations.
   - RP/Chemistry, if available.
   - Engineering, if available.

   6.3 **RECORD** Time Since Shutdown under all conditions in hours value in Table 6-1, Block 3.

7. **DETERMINE** if core is degraded based on plant conditions.

   7.1 **REFER** to Attachment 8 for guidance on degraded core.©[4]

   7.2 **CONSULT** with following to coordinate answer:©[5]
   - Operations.
   - RP/Chemistry, if available.
   - Engineering, if available.

   7.3 IF core degraded,
   THEN **OBTAIN** Iodine to Noble Gas ratio from Table 6-6.

      7.3.1 **RECORD** Iodine to Noble Gas ratio value in Table 6-1, Block 4.

   7.4 IF core not degraded,
   THEN **RECORD** 1.86E-07 in Table 6-1, Block 4.

8. **DETERMINE** Noble Gas Energy Factor (MeV/dis) using Table 6-7 based on time since shutdown in hours.

   8.1 **RECORD** Noble Gas Energy Factor (MeV/dis) value in Table 6-1, Block 5.

NOTE – MET Tower instrumentation from different trains may be used for each elevation or parameter when performing dose assessment with "A" being preferred train. If 100 meter MET tower is out of service or unavailable, PMIS will default to next highest tower if elevated release. Default is 60M.

9. **OBTAIN** 15 minute average wind speed in miles per hour (mph).

   9.1 IF release from ERP,
   THEN **USE** data in following preferred order for elevated release:

      9.1.1 PMIS 100 meter level.

      9.1.2 PMIS 60 meter level.

      9.1.3 **CONTACT** NWS at 402-359-4381 or refer to telephone directory and **OBTAIN** average wind speed estimate for 100 meter elevation.

      9.1.4 Historical default wind speed value of 13 mph.

   9.2 IF release from any other source,
   THEN **USE** data in following preferred order for ground level release:

      9.2.1 PMIS 10 meter level.

      9.2.2 PMIS 60 meter level.

      9.2.3 **CONTACT** NWS at 402-359-4381 or refer to telephone directory and **OBTAIN** average wind speed estimate for 10 meter elevation.

      9.2.4 Historical default wind speed value of 8 mph.

   9.3 **RECORD** 15 minute average wind speed in Table 6-1, Block 6.

10. **OBTAIN** wind direction (degree from).

10.1 IF release source from ERP,
THEN **USE** data in following preferred order:

10.1.1 PMIS 100 meter level.

10.1.2 PMIS 60 meter level.

10.1.3 **CONTACT** NWS at 402-359-4381 or refer to telephone directory and **OBTAIN** average wind direction estimate at CNS (degree from) at 100 meter elevation.

10.2 IF release source from any other source,
THEN **USE** data in following preferred order:

10.2.1 PMIS 10 meter level.

10.2.2 PMIS 60 meter level.

10.2.3 **CONTACT** NWS at 402-359-4381 or **REFER** to telephone directory and **OBTAIN** average wind direction estimate at CNS (degree from) for 10 meter elevation.

10.3 **RECORD** wind direction (degree from) in Table 6-1, Block 7.

11. **DETERMINE** atmospheric stability class ("A" through "G").

11.1 IF release from ERP,
THEN **USE** data in following preferred order:

11.1.1 Direct PMIS stability class readout.

11.1.2 PMIS data below and Table 5-2.

100 meter (°F) ______ minus 10 meter (°F) ______ equals delta-T ______

Table 5-2

Delta-T °F	≤ -3.08	> -3.08 to ≤ -2.75	> -2.75 to ≤ -2.43	> -2.43 to ≤ -0.81	> -0.81 to ≤ 2.43	> 2.43 to ≤ 6.48	> 6.48
Stability Class	A	B	C	D	E	F	G

11.1.3 PMIS data below and Table 5-3.

60 meter (°F) ______ minus 10 meter (°F) ______ = delta-T ________

Table 5-3							
Delta-T °F	≤ -1.71	> -1.71 to ≤ -1.46	> -1.46 to ≤ -1.35	> -1.35 to ≤ -0.45	> -0.45 to ≤ 1.35	> 1.35 to ≤ 3.6	> 3.6
Stability Class	A	B	C	D	E	F	G

11.1.4 PMIS 100 meter sigma-theta data below and Table 5-4.

11.1.5 PMIS 60 meter sigma-theta data below and Table 5-4.

11.1.6 PMIS 10 meter sigma-theta data below and Table 5-4.

Sigma-theta from 100 M, 60 M, or 10 M: ____________

Table 5-4							
Sigma-Theta	≥ 22.5	< 22.5 to ≥ 17.5	< 17.5 to ≥ 12.5	< 12.5 to ≥ 7.5	< 7.5 to ≥ 3.8	< 3.8 to ≥ 2.1	< 2.1
Stability Class	A	B	C	D	E	F	G

11.1.7 NWS data below and Table 5-5.

11.1.7.1 **CONTACT** NWS at 402-359-4381 or refer to telephone directory.

100 meter (°F) ______ minus 10 meter (°F) ______ = delta-T ________

Table 5-5							
Delta-T °F	≤ -3.08	> -3.08 to ≤ -2.75	> -2.75 to ≤ -2.43	> -2.43 to ≤ -0.81	> -0.81 to ≤ 2.43	> 2.43 to ≤ 6.48	> 6.48
Stability Class	A	B	C	D	E	F	G

11.1.8 **USE** historical default "D".

11.2 IF release from any other source,
THEN **USE** data in following preferred order:

11.2.1 Direct PMIS stability class readout.

11.2.2 PMIS data below and Table 5-6.

60 meter (°F) ______ minus 10 meter (°F) ______ = delta-T __________

Table 5-6

Delta-T °F	≤ -1.71	> -1.71 to ≤ -1.46	> -1.46 to ≤ -1.35	> -1.35 to ≤ -0.45	> -0.45 to ≤ 1.35	> 1.35 to ≤ 3.6	> 3.6
Stability Class	A	B	C	D	E	F	G

11.2.3 PMIS data below and Table 5-7.

100 meter (°F) ______ minus 10 meter (°F) ______ = delta-T __________

Table 5-7

Delta-T °F	≤ -3.08	> -3.08 to ≤ -2.75	> -2.75 to ≤ -2.43	> -2.43 to ≤ -0.81	> -0.81 to ≤ 2.43	> 2.43 to ≤ 6.48	> 6.48
Stability Class	A	B	C	D	E	F	G

11.2.4 PMIS 10 meter sigma-theta data below and Table 5-8.

11.2.5 PMIS 60 meter sigma-theta data below and Table 5-8.

11.2.6 PMIS 100 meter sigma-theta data below and Table 5-8.

Sigma-theta from 10 M, 60 M, or 100M: ______________

Table 5-8

Sigma-Theta	≥ 22.5	< 22.5 to ≥ 17.5	< 17.5 to ≥ 12.5	< 12.5 to ≥ 7.5	< 7.5 to ≥ 3.8	< 3.8 to ≥ 2.1	< 2.1
Stability Class	A	B	C	D	E	F	G

11.2.7 NWS data below and Table 5-9.

11.2.7.1 **CONTACT** NWS at 402-359-4381 or refer to telephone directory.

60 meter (°F) _______ minus 10 meter (°F) ______ = delta-T __________

Table 5-9

Delta-T °F	≤ -1.71	> -1.71 to ≤ -1.46	> -1.46 to ≤ -1.35	> -1.35 to ≤ -0.45	> -0.45 to ≤ 1.35	> 1.35 to ≤ 3.6	> 3.6
Stability Class	A	B	C	D	E	F	G

11.2.8 **USE** historical default "D".

11.3 **RECORD** stability class value in Table 6-1, Block 8.

12. **DETERMINE** if release pathway can credit Reactor Building for plate-out:

12.1 **CONSULT** with following to coordinate answer:

- Operations.
- RP/Chemistry, if available.
- Engineering, if available.

12.2 Use Table 5-10 to **SELECT** appropriate entry for release pathway based on release path and plant conditions.

Table 5-10		
RELEASE SOURCE	RELEASE PATH	ENTER
ERP	Reactor Bldg through SGT without Hard Pipe Vent	Yes-0.5
ERP	Primary Containment through Hard Pipe Vent	No-1.0
ERP	Gland Seal Steam, Main Condenser through Air Removal, and Off-Gas	No-1.0
Reactor Bldg	Reactor Bldg HVAC or direct ground level from blowout panel or other opening	Yes-0.5
Turbine Bldg	TB HVAC or direct ground level from blowout panel or other opening	No-1.0
Turbine Bldg	TB HVAC plus Reactor Building via Reactor Building-TB blowout panel	Yes-0.5
RW/ARW Bldg	RW/ARW HVAC or other direct ground level opening	No-1.0

12.3 IF NO indicating release path bypasses Reactor Building, THEN **RECORD** 1.0 in Table 6-1, Block 9.

12.4 IF YES indicating release path into or through Reactor Building, THEN **RECORD** 0.5 in Table 6-1, Block 9.

13. **OBTAIN** TEDE Noble Gas Dose Conversion Factor from Table 6-8.

13.1 **RECORD** TEDE Noble Gas Dose Conversion Factor value in Table 6-2, Block 10.

14. **OBTAIN** TEDE Iodine Dose Conversion Factor from Table 6-8.

14.1 **RECORD** TEDE Iodine Dose Conversion Factor value in Table 6-2, Block 11.

15. **OBTAIN** CDE Iodine Dose Conversion Factor from Table 6-8.

15.1 **RECORD** CDE Iodine Dose Conversion Factor value in Table 6-2, Block 12.

16. **DETERMINE** mixing factor (χ/Q) value for non-centerline receptor location.

16.1 **RECORD** receptor location including ID and distance from release source at top of Attachment 6.

NOTE – Overlays are available in TSC and EOF for both elevated and ground level releases for each stability class. Ground level isopleths are used for all releases which are not from ERP.©[10]

16.2 **OBTAIN** proper χ/Q isopleth overlay based on stability class and release point.©[10]

NOTE – Preferred map is map with sectors, radii, and wind direction labeled, located in TSC and EOF.

16.3 **PLACE** isopleth overlay on 10 mile Emergency Planning Zone map scaled to 1 inch per mile.

16.4 **ENSURE** isopleth overlay is oriented so all of following are met:

- Centerline of isopleth is over wind direction radius.
- Open end of isopleth is downwind.
- Isopleth asterisk zero directly over zero point on map representing plant.

16.5 **MARK** desired receptor location on isopleth.

NOTE 1 – All χ/Qs values have negative exponents.

NOTE 2 – Interpolation between isopleth segment χ/Q values may be used to increase accuracy when receptor location lies between isopleth segment lines.

16.6 On Isopleth, **DETERMINE** segment containing receptor location.

16.6.1 **RECORD** Isopleth segment associated containing receptor location in Table 5-11, Block 1.

16.7 **USE** χ/Q VALUES table, ISOPLETHS column, to determine χ/Q value associated with isopleth segment containing receptor location recorded in Table 5-11, Block 1.

16.7.1 **RECORD** χ/Q value associated with isopleth segment containing receptor location Table 5-11, Block 2.

16.8 On Isopleth, **DETERMINE** segment associated with plume centerline at same distance from release source as receptor location.

16.8.1 **RECORD** Isopleth Segment at Plume Centerline at same distance as Receptor location in Table 5-11, Block 3.

16.9 **USE** χ/Q VALUES table, ISOPLETHS column, to determine χ/Q value associated with segment associated with plume centerline.

16.9.1 **RECORD** χ/Q value associated with plume centerline in Table 5-11, Block 4.

16.10 **DETERMINE** non-centerline receptor location adjustment ratio:

16.10.1 **DIVIDE** Table 5-11, Block 2 by Block 4.

16.10.2 **RECORD** non-centerline receptor location adjustment ratio in Table 5-11, Block 5.

Table 5-11

(1) Isopleth Segment at Receptor Location (A-Q)	(2) χ/Q Value for Isopleth Segment at Receptor Location	(3) Isopleth Segment at Plume Centerline at same Distance as Receptor Location (A-Q)	(4) χ/Q Value for Isopleth Segment at Plume Centerline	(5) Non-Centerline Receptor Location Segment Adjustment Ratio (Block 2) ÷ (Block 4) (<1.0)

16.11 **USE** χ/Q VALUES table, CENTERLINE column, to determine plume centerline χ/Q value associated receptor distance from release source.

16.11.1 IF receptor location lies between mile distances, THEN **PERFORM** interpolation to determine more accurate adjustment.

- Example: Assuming receptor location is 7.5 miles and plume centerline χ/Q values at 7 miles is 8.31E-08 and 6.60E-08 at 8 miles. Therefore, interpolated plume centerline χ/Q value for 7.5 miles would be 7.455E-08 or rounded to 7.46E-08.

16.11.2 **RECORD** χ/Q plume centerline value associated with receptor distance in Table 5-12, Block 1.

16.12 **RECORD** non-centerline receptor location adjustment ratio from Table 5-11, Block 5, in Table 5-12, Block 2.

16.13 **DETERMINE** Adjusted Mixing Factor at receptor location:

16.13.1 **MULTIPLY** Table 5-12, Block 1 by Block 2.

16.13.2 **RECORD** Adjusted Mixing Factor at receptor location in Table 5-12, Block 3.

Table 5-12		
(1) χ/Q Value for Receptor Location Distance per Isopleth Centerline	(2) Non-Centerline Receptor Location Segment Adjustment Ratio (Block 5 of Table 5-11)	(3) Adjusted Mixing Factor at Receptor Location (Block 1 x Block 2)

16.14 **RECORD** Adjusted Mixing Factor at Receptor Location from Table 5-12, Block 3, into Table 6-2, Block 13.

17. **CALCULATE** TEDE dose rate (rem/hr).

17.1 **MULTIPLY** Table 6-1, Block 1 by Block 5, by Table 6-2, Block 10, to obtain Noble Gas portion.

17.2 **MULTIPLY** Table 6-1, Block 1 by Block 2 by Block 4 by Block 9, by Table 6-2, Block 11, to obtain Iodine portion.

17.3 **ADD** Noble Gas results from Step 17.1 and Iodine results from Step 17.2.

17.4 **DIVIDE** results of Step 17.3 by Table 6-1, Block 6, to complete TEDE Sub-Calculation.

17.5 **MULTIPLY** TEDE Sub-Calculation results from Step 17.4 by Table 6-2, Block 13.

NOTE – Hand calculated dose assessment projections shall not be corrected downward based on field samples.

17.6 **OBTAIN** Field Sample Correction Factor (CF) results from Table 7-1, Column 6.

17.6.1 IF Field Sample CF results unavailable or less than 1.0, THEN **RECORD** 1.0 and **PROCEED** to Step 17.7.

17.6.2 IF Field Sample CF results greater than 1.0, THEN **PERFORM** following:

17.6.2.1 **RECORD** Field Sample CF value into upper Table 6-3, Block 14.

17.6.2.2 **MULTIPLY** results of Step 17.4 by Field Sample CF to raise TEDE dose rate assessment.

17.7 **RECORD** TEDE dose rate (rem/hr) value in Table 6-3, Block 14.

18. **ESTIMATE** duration of release in hours.

18.1 **CONSULT** with following to coordinate answer:

- Operations.
- RP/Chemistry, if available.
- Engineering, if available.

18.2 IF duration of release unknown, THEN **USE** 4 hours as default.

18.3 **RECORD** duration of release in Table 6-4, Block 15.

19. **CALCULATE** integrated TEDE dose (rem).

19.1 **MULTIPLY** Table 6-3, Block 14, by Table 6-4, Block 15.

19.2 **RECORD** TEDE dose (rem) value in Table 6-4, Block 16.

20. IF higher level emergency classification is discovered after calculating Block 16 value, THEN immediately **INFORM** ED or RCM.

21. **CALCULATE** CDE dose rate (rem/hr).

21.1 **MULTIPLY** Table 6-1, Block 1 by Block 2 by Block 4 by Block 9, and then by Table 6-2, Block 12.

21.2 **DIVIDE** results by Table 6-1, Block 6, to complete CEDE Sub-Calculation.

21.3 **MULTIPLY** results from Step 21.2 by Table 6-2, Block 13.

NOTE – Hand Calculated dose assessment projections shall not be corrected downward based on field samples.

21.4 **OBTAIN** Field Sample Correction Factor (CF) results from Table 7-1, Column 6.

21.4.1 IF Field Sample CF results unavailable or less than 1.0, THEN **RECORD** 1.0 and **PROCEED** to Step 21.5.

21.4.2 IF Field Sample CF results greater than 1.0, THEN **PERFORM** following:

21.4.2.1 **RECORD** Field Sample CF value into Table 6-5, upper Block 17.

21.4.2.2 **MULTIPLY** results of Step 21.3 by Field Sample CF to raise CDE dose rate assessment.

21.5 **RECORD** CDE dose rate (rem/hr) value in Table 6-5, Block 17.

22. **CALCULATE** CDE dose (rem).

22.1 **MULTIPLY** Table 6-4, Block 15, by Table 6-5, Block 17.

22.2 **RECORD** CEDE dose (rem) value in Table 6-5, Block 18.

23. **RECORD** name, time, and date at bottom of Attachment 6.

24. IF Attachment 6 was completed electronically, THEN **PRINT** completed Attachment 6.

25. IF multiple release points are identified,
THEN **DETERMINE** if summation of various releases could result in PAR change or change in emergency classification level.

25.1 **PERFORM** dose assessment for each release point by repeating Steps 2 through 24.

25.2 **OBTAIN** copy of Excel spreadsheet from EP webpage under Dose Assessment tab using business computer.

25.3 **SAVE** spreadsheet to user directory of business computer.

25.4 **DEPRESS** Clear Input Screen button on Input tab.

25.5 **RECORD** following from Attachment 6 on Input tab in proper column based on release point:

- Release Rate from Table 6-1, Block 1.
- Integrated TEDE dose from Table 6-4, Block 16, and Integrated CEDE dose from Table 6-5, Block 18.
- TEDE dose rate from Table 6-3, Block 14, and CED dose rate from Table 6-5, Block 17.

25.6 **RECORD** wind directions for Elevated Release Point and Ground Level, as appropriate, for multiple releases.

25.7 WHEN all identified release points are entered,
THEN **DEPRESS** Print Package button.

25.7.1 **ENSURE** all pages are printed to ensure accuracy during reviews.

25.7.2 **ENSURE** times are recorded on appropriate pages.

25.8 IF spreadsheet cannot be obtained, saved, or used,
THEN **PERFORM** manual summation of dose assessments based on each sector and distance utilizing Attachment 10 or similar data sheet.

26. **NOTIFY** Radiological Assessment Supervisor (RAS) and RCM, dose projection(s) and multi-source releases, if required, are complete.

    26.1 RAS or RCM **REVIEW** dose projection(s).

    26.2 RCM **DETERMINE** if emergency classification or PAR is affected by radiological effluent (dose rate or integrated dose to a member of public) calculated at or beyond 1 mile.

        26.2.1 RCM **REFER** to Technical Bases Effluent Monitor Classification Thresholds Table in EPIP 5.7.1.

        26.2.2 RCM **RECOMMEND** emergency classification or PAR change to ED.

27. <u>WHEN</u> Field Monitoring Team reports air sample results at receptor location, <u>THEN</u> **PERFORM** following to Attachment 7:©[3]

    27.1 **COMPARE** direct field sample readings to hand calculated non-centerline receptor location dose projection results of Attachment 6.

    27.2 **APPLY** Field Sample Correction Factor, as applicable, to adjust hand calculated non-centerline dose projection accordingly.

28. **NOTIFY** RCM that field sample comparison to hand calculated dose projections is complete.

29. RCM **RECOMMEND** if change to PARs necessary.

ATTACHMENT 6 HAND CALCULATED DOSE PROJECTION (NON-CENTERLINE) WORKSHEET (INFORMATION USE)

Receptor Location (Receptor ID and distance from release source): ______ / ______ Miles

Table 6-1

(1) Noble Gas Release Rate from KAMAN or 5.7.16 (μCi/Sec)	(2) Release Path through SGT?	(3) Time Since Shutdown (hours)	(4) Iodine/Noble Gas Ratio (from Table 6-6)	(5) Noble Gas Energy Factor (MeV/dis) (from Table 6-7)	(6) Wind Speed (mph)	(7) Wind Direction (° from)	(8) Stability Class	(9) Release through Reactor Building?

Table 6-2

Conversion Factors (from Table 6-8)		(13) Adjusted Mixing Factor at receptor
TEDE Noble Gas (10)		
TEDE Iodine (11)		
CDE Iodine (12)		

Table 6-3

(14) TEDE Dose Rate (rem/hr)
$\frac{[(\text{Block 1})(\text{Block 5})(\text{Block 10})]+[(\text{Block 1})(\text{Block 2})(\text{Block 4})(\text{Block 9})(\text{Block 11})]}{(\text{Block 6})}$ × (Block 13) × Field Sample CF: (Table 7-1)
(14) rem/hr

Table 6-4

(15) Duration of Release (hours)	(16) TEDE Dose (rem) (Block 14) x (Block 15)

Table 6-5

(17) CDE Dose Rate (rem/hr) $\frac{(\text{Block 1})(\text{Block 2})(\text{Block 4})(\text{Block 9})(\text{Block 12})]}{(\text{Block 6})}$ × (Block 13) × Field Sample CF: (Table 7-1)	(18) CDE Dose (rem) (Block 15) x (Block 17)
(17) rem/hr	

Name/Time/Date: ______ / ______ / ______

Table 6-6: Iodine To Noble Gas Ratio vs. Time Since Shutdown		
TIME SINCE SHUTDOWN (hrs)	IODINE/NOBLE GAS RATIO	
	NON-DEGRADED CORE	DEGRADED CORE
t < 1	1.86 E-7	2.71 E-1
1 ≤ t < 2		3.57 E-1
2 ≤ t < 4		3.41 E-1
4 ≤ t < 10		2.81 E-1
10 ≤ t < 30		2.30 E-1
30 ≤ t < 100		1.65 E-1
100 ≤ t		1.40 E-1

Table 6-7: Noble Gas Energy Factors	
TIME SINCE SHUTDOWN (hrs)	ENERGY FACTOR (MeV/dis)
t < 1	0.75
1 ≤ t < 2	0.60
2 ≤ t < 4	0.40
4 ≤ t < 10	0.25
10 ≤ t < 30	0.15
30 ≤ t < 100	0.09
100 ≤ t	0.07

Table 6-8: Dose Conversion Factors		
	NON-DEGRADED CORE	DEGRADED CORE
TEDE Noble Gas	1.48 E-3	9.19 E-4
TEDE Iodine	8.77 E-2	2.98 E-2
CDE Iodine	2.04 E 0	4.96 E-1

ATTACHMENT 7 CORRELATING OFF-SITE SAMPLE RESULTS WITH HAND CALCULATED DOSE PROJECTIONS (NON-CENTERLINE) (REFERENCE USE)©3

1. **COMPLETE** Field Monitoring Team sample information in Table 7-1.
   - **RECORD** receptor location identification (ID) and downwind distance (miles) from site in Column (1).
   - **RECORD** field sample downwind sector (A-Q) in Column (2).
   - **RECORD** field sample time in military format (0000 hrs) in Column (3).
   - **RECORD** TEDE dose Rate (rem/hr) in Column (4) from direct field sample.
2. In Table 7-1, Column (5), **RECORD** hand calculated (non-centerline) projected TEDE dose rate (rem/hr) from Table 6-3, Block 14, for receptor location selected in Column (1).
3. **DIVIDE** Column (4) by Column (5) and **RECORD** Field Sample Correction Factor (CF) in Table 7-1, Column (6).
4. **RECORD** Initials, Date, and Time of comparison on Table 7-1.
5. **PROVIDE** Attachment 7 to RAS or RCM for review.
6. IF Field Sample CF in Table 7-1, Column (6), is less than 1.0, THEN **STOP**. No field sample correction of non-centerline dose projection required for this receptor location.
7. IF Field Sample CF in Table 7-1, Column (6), is greater than 1.0 indicating field sample results are greater than hand calculated dose rate projections, THEN **PERFORM** following:

   7.1 RAS or RCM **APPROVE** Field Sample CF results in Table 7-1, Column (6), to adjust hand calculated dose assessment.

   7.1.1 **APPLY** approved Field Sample CF during subsequent performance of Attachment 5.

   7.2 RCM **DETERMINE** if emergency classification or PAR is affected by radiological effluent (dose rate or integrated dose to a member of public) calculated at or beyond 1 mile.

   7.2.1 RCM **REFER** to Technical Bases Effluent Monitor Classification Thresholds Table in EPIP 5.7.1.

   7.2.2 RCM **RECOMMEND** emergency classification or PAR change to ED.

Table 7-1

(1) Receptor/ Sample Location (ID/miles)	(2) Receptor/ Sample Location Sector (A-Q)	(3) Field Sample Time (0000 hrs)	(4) Direct Field Sample Dose Rate Reading at Receptor Location (rem/hr)	(5) Hand Calculated (Non-Centerline) Projected Dose Rate (rem/hr)	(6) Field Sample Correction Factor (CF)* (Column 4 ÷ Column 5)	Initials/Date/Time
						____/____/____
						____/____/____
						____/____/____
						____/____/____

* Field Sample Correction Factor (CF) results greater than 1.0 indicates hand calculation results are non-conservative. This condition is resolved by application of Field Sample CF during subsequent Hand Calculated dose projections.

ATTACHMENT 8 DEGRADED CORE DETERMINATION GUIDE (INFORMATION USE)

NOTE – Response to Degraded Core question has a significant impact on Iodine to Noble Gas ratio and resultant dose projection calculations.©[5]

1. Core is considered Degraded if any of following conditions are met or were met and have subsequently dropped below threshold:©[4]

   1.1 Primary Coolant Activity greater than 300 μCi/gm Dose Equivalent I-131.©[9]

   1.2 DW Rad Monitor reading greater than 2500 rem/hr.

   1.3 Primary Containment flooding is required due to any of following:

   1.3.1 RPV water level cannot be restored and maintained greater than -183 inches.

   1.3.2 RPV water level cannot be restored and maintained greater than or equal to -209 inches and no Core Spray Subsystem flow can be restored and maintained greater than or equal to 4750 gpm.

   1.3.3 RPV water level cannot be determined and core damage is occurring.

   1.4 Any condition in opinion of Emergency Director that indicates loss of fuel clad barrier. Examples:

   - SJAE monitor reading 15,000 mrem/hr.
   - Non-LOCA with DW Rad Monitor reading greater than 115 rem/hr.
   - Main Steam Line Radiation Monitor readings greater than or equal to Hi-Hi Alarm Setpoint.

ATTACHMENT 9 CORE DAMAGE ESTIMATION USING IN-CONTAINMENT HI-RANGE RADIATION MONITORS (REFERENCE USE)©[1]

NOTE – Attachment 9 is only used for core damage estimates where in-containment radiation monitors are exposed to coolant or steam (i.e., only for primary containment LOCA situations). For other accident sequences, Post-Accident Sampling System (PASS) may be used, as required, to obtain Reactor Coolant System (RCS) sample.©[7]

1. IF both following conditions met:©[7]
   - Loss of coolant accident in primary containment has not occurred.
   - Post-Accident Sampling System (PASS) is available.

   THEN Chemistry/RP Coordinator or designee **PERFORM** following:

   1.1 **EXIT** this attachment.

   1.2 **OBTAIN** a Reactor Coolant System (RCS) sample using PASS or other means as required.

   1.3 **PERFORM** core damage assessment using Core Damage Assessment Program (CORDAM).

NOTE 1 – Release from core may bypass containment, be retained in primary system, or not uniformly mixed. Therefore, a low containment radiation reading does not guarantee a lack of core damage. Levels of damage indicated by value in Table 9-1 are considered minimum levels unless there are inconsistent monitor readings.

NOTE 2 – Inconsistent monitor readings may be due to uneven mixing in containment (e.g., steam rising to top of dome). It may take hours for uniform mixing.

1. IF Loss of Coolant Accident in primary containment has occurred, THEN Chemistry/RP Coordinator or designee **PERFORM** following:

   1.1 **OBTAIN** highest in-containment hi-range radiation monitor reading from RMA-RM-40A (B), DRYWELL RAD MONITOR.

   1.1.1 **RECORD** value in Table 9-1, Block 1.

   1.2 **CALCULATE** Core Melt Fraction:

   1.2.1 **DIVIDE** Table 9-1, Block 1 by Block 2.

   1.2.2 **RECORD** Core Melt Fraction results in Table 9-1, Block 3.

1.3 **CALCULATE** Percent Core Melt:

1.3.1 **MULTIPLY** Table 9-1, Block 3, by 100.

1.3.2 **RECORD** Percent Core Melt results in Table 9-1, Block 4.

1.4 **CALCULATE** Percent Clad Failure:

1.4.1 **MULTIPLY** Table 9-1, Block 4, by 10.

1.4.2 **RECORD** Percent Clad Failure results in Table 9-1, Block 5.

1.5 **REPORT** results of core damage estimate (Blocks 4 and 5) to RCM and TSC Director.

Table 9-1

(1) HIGHEST DRYWELL RAD MONITOR READING (RMA-RM-40A,B)	(2) 100% CORE MELT FACTOR	(3) CORE MELT FRACTION (1) ÷ (2)	(4) PERCENT CORE MELT (3) x 100	(5) PERCENT CLAD FAILURE (4) x 10
	2.44E+6			

Name/Time/Date: ______________________ / __________ / __________

ATTACHMENT 10 MULTI-SOURCE RELEASE SUMMATION EXAMPLE (INFORMATION USE)

Elevated Wind Direction	**0**
Ground Level Wind Direction	**0**

Release Rate uCi/sec					
Integrated Dose TEDE (Rem)	**ERP**	**RX**	**ARW**	**TG**	**Other**
Mile					
1					
2					
5					
10					

Integrated Dose CDE (Rem)

Mile					
1					
2					
5					
10					

Dose Rate TEDE (Rem/hr)

Mile					
1					
2					
5					
10					

Dose Rate CDE (Rem/hr)

Mile					
1					
2					
5					
10					

Time:

Integrated Dose TEDE (Rem)

Mile	Sector A	Sector B	Sector C	Sector D	Sector E	Sector F	Sector G	Sector H	Sector J	Sector K	Sector L	Sector M	Sector N	Sector P	Sector Q	Sector R
1																
2																
5																
10																

Release Rate (uCi/sec)

Integrated Dose CDE (Rem)

Mile	Sector A	Sector B	Sector C	Sector D	Sector E	Sector F	Sector G	Sector H	Sector J	Sector K	Sector L	Sector M	Sector N	Sector P	Sector Q	Sector R
1																
2																
5																
10																

Dose Rate TEDE (Rem/hr)

Mile	Sector A	Sector B	Sector C	Sector D	Sector E	Sector F	Sector G	Sector H	Sector J	Sector K	Sector L	Sector M	Sector N	Sector P	Sector Q	Sector R
1																
2																
5																
10																

Dose Rate CDE (Rem/hr)

Mile	Sector A	Sector B	Sector C	Sector D	Sector E	Sector F	Sector G	Sector H	Sector J	Sector K	Sector L	Sector M	Sector N	Sector P	Sector Q	Sector R
1																
2																
5																
10																

ATTACHMENT 11 TRANSIT TIMES AND EFFECTIVE AGES OF NOBLE GASES AT RECEPTOR SITES (INFORMATION USE)

1. Effective Age is defined as time elapsed (hrs) since shutdown. For off-site locations, effective age of isotopic mixture may be obtained through summarizing following components:
   - Effective age at time of release onset.
   - Transit time from release point to receptor site.

2. CALCULATION OF TRANSIT TIME FROM RELEASE POINT TO RECEPTOR LOCATION
   - Estimate downwind distance (miles) to receptor location.
   - Divide distance in miles by 100 meter level (if release from ERP) or 10 meter (if release not from ERP) wind speed (mph) to determine plume transit time.

(1) RECEPTOR SITE DOWNWIND DISTANCE (miles)	(2) 100 METER or 10 METER LEVEL WIND SPEED (as appropriate) (mph)	(3) PLUME TRANSIT TIME (hrs) (1) ÷ (2)

DETERMINATION OF EFFECTIVE AGES AT RECEPTOR SITES

(1) EFFECTIVE AGE OF MIXTURE AT TIME OF RELEASE ONSET (hrs)	(2) TRANSIT TIME FROM RELEASE POINT TO RECEPTOR LOCATION (hrs)	(3) EFFECTIVE AGE OF ISOTOPIC MIXTURE AT RECEPTOR LOCATION (hrs) (1) + (2)

Name/Time/Date: ____________________ / ________ / ________

ATTACHMENT 12 PMIS SYSTEM ACCESS AND USE (INFORMATION USE)

1. PLANT MANAGEMENT INFORMATION SYSTEM (PMIS)

   1.1 The PMIS System (PMIS) is a set of programs and hardware provided by NPPD that make use of VMS functions and additional peripherals (Data Concentrators) which provides access to plant parameters.

2. PMIS COMPUTERS

   2.1 PMIS computers share a common set of peripherals (disk drives, terminals, etc.) and software common to that computers security level.

3. VMS OPERATING SYSTEM

   3.1 The Virtual Memory System (VMS) Operating System is the host operating system for the PMIS computers. It is a set of programs that interface with computer hardware and peripherals, and allows computers to recognize and process commands.

4. PMIS MODES

   4.1 PMIS has three operational modes, Primary, Primary/Backup, and Backup, and will operate on either computer in one of these three modes. A computer with PMIS operating in either Primary or Primary/Backup Mode is referred to as the Primary System and one with PMIS operating in Backup Mode is referred to as Backup System.

   4.2 The Primary and Primary/Backup Modes provide full PMIS capabilities, consisting (in part) of data acquisition and conversion, data display, data archiving, alarm processing, self-monitoring, and many other functions that perform specialized calculations and displays.

   4.3 The Backup Mode monitors Primary System, transfers information necessary to keep Backup System files and tables up-to-date, and automatically changes to Primary Mode when a loss of Primary System is detected (referred to as a FAILOVER). Although many functions are available on Backup System, their use is discouraged because lack of real-time data results in display of inaccurate information (CNS-DOSE is an exception).

5. PMIS ACCESS

   5.1 Access to PMIS is gained through various video display terminals and printers, including color graphic Information Display Terminals (IDTs) dedicated exclusively for PMIS access in Control Room, TSC, and EOF.

5.2 The IDTs and printers are selectively connected to either computer through networks controlled by PMIS.

6. SCREEN FORMAT

6.1 When a terminal is under control of PMIS (instead of VMS), the screen display will be in a standard format consisting of four areas, OCA, GGDA, SSA, and FKA.

6.2 The OCA (Operator Communication Area) consists of the top two lines on the screen. This area is generally used to prompt for and receives user inputs and display advisory and warning messages. In addition, some displays that require only one or two lines of screen use OCA for display. Also (though technically not part of OCA), current date and time (updated once a second) is displayed at right side of screen on lines 1 and 2.

6.3 The GGDA (General and Graphic Display Area) consists of lines 4 through 47 and is used for most displays. In addition, some displays (chiefly functions requiring significant editing) also prompt for and receive user inputs in the GGDA.

6.4 The SSA (SPDS Status Area) consists of lines 45 through 48 and contain four boxes that represent (by color code) the status of the SPDS (Safety Parameter Display System), which is a software system that monitors selected plant parameters and determines overall plant safety status.

6.5 The FKA (Function Key Area) consists of bottom two (50 and 51) lines of screen. FKA is used to indicate which of the definable function keys are enabled. It also indicates which mode PMIS is in, Plant Mode, and whether or not a PMIS "event" has occurred.

7. SCREEN-COPY FUNCTION

7.1 The screen-copy function, which is activated by depressing HARD COPY key, provides full screen reproduction in color on a printer located in same general area as terminal.

8. PRINTER

8.1 Printers are connected to a specific computer and are generally accessed when a "...PRINT..." option is selected and a "logical name" is entered.

9. IDE FIELD

9.1 User input to PMIS Programs is through an open IDE (Interactive Data Entry) field on terminal. An open IDE field is denoted by a yellow box that appears in OCA or GGDA area. Anything typed on keyboard will be echoed in box. Erasing or back-spacing is accomplished with DEL key. All entries into an IDE field must be terminated by depressing ENTER key unless field is overfilled or a function key is depressed (terminal automatically adds a carriage return character in those cases).

10. TURN-ON-CODE

10.1 Turn-On-Code (TOC) is the mechanism by which commands are issued to PMIS. This is a one to eight character code which is interpreted by PMIS and a corresponding command is issued.

11. PMIS DATABASE

11.1 All plant parameters (or additional data based on plant or PMIS parameters) that are processed by PMIS SYSTEM are defined in PMIS DATABASE, which is a file that specifies origin of data, frequency at which it is processed, type of processing to be performed, etc. Each parameter is referred to as a "point" and is identified by a one to eight character name or POINT-ID (PID).

12. PMIS DATA PROCESSING

12.1 Some PMIS points are processed by scanning plant sensors (through Data Concentrator) while others are calculated based on values of previously processed points or PMIS parameters. All point values are then assigned a quality code stored in Current Value Table (CVT).

12.2 Data in CVT is considered to be "real-time" and representative of current plant and system conditions.

12.3 At regular intervals (and other special circumstances), point values are also stored in an Archive File, which provides ~ 24 hours of on-line historical information.

13. PMIS DATA ACCESS

13.1 All point values in the CVT and Archive File are accessed by the POINT-ID.

14. QUALITY CODES

14.1 Quality Code, assigned when point values are assigned, represents general status and "health" of point, and determines how it is used by PMIS Programs. The following is a list of PMIS quality codes and related information.

CODE	DESCRIPTION	COLOR	HEALTH
UNK	Value unknown - not yet processed	White	Bad
DEL	Processing has been disabled	Magenta	Bad
NCAL	Value cannot be calculated	Magenta	Bad
INVL	Data concentrator error	Magenta	Bad
RDER	Data concentrator error	Magenta	Bad
OIC	Data concentrator error	Magenta	Bad
BAD	Outside instrument range	Magenta	Bad
STAG	Point failed stagnation check	Magenta	Bad
UDEF	Undefined (spare)	Magenta	Bad
REDU	Fails redundant point check	Magenta	Bad
HALM	Above high alarm limit	Red	Good
LALM	Below low alarm limit	Red	Good
HWRN	Above high warning limit	Yellow	Good
LWRN	Below low warning limit	Yellow	Good
ALM	State/Change-of-State alarm	Red	Good
SUB	Value has been substituted	Blue	Good
DALM	Alarm checking has been disabled	Green	Good
INHB	Alarm inhibited by cut-out point	Green	Good
GOOD	Passes all other checks	Green	Good

14.2 Not listed above is quality code OSUB (Operator Substituted), which is treated same as SUB, and indicates that value was substituted within that program. OSUB is not used in CVT.

15. PMIS LOGIN

15.1 IF windows desktop is being displayed,
THEN **START** IDT software.

15.2 IF Connection and Data Server Information dialog box is displayed,
THEN **DEPRESS** Attempt Connection button.

15.3 IF current date and time is displayed in OCA and is being updated about once a second,
THEN **PERFORM** following:

NOTE – Password information will not be displayed for a CDA IDT as it is entered automatically.

15.3.1 IF "ENTER PASSWORD..." is displayed on line 2 for a CDA IDT,
THEN **RE-BOOT** IDT.

15.3.2 IF IDT is business computer,
THEN **ENTER** designated password.

15.3.3 IF "SELECT FUNC. KEY OR TURNS ON CODE..." and an open IDE field is displayed on line 2,
THEN IDT is logged into PMIS; no further action is necessary.

15.3.4 IF display is operating (such as DOSE program),
THEN **DEPRESS** CANC key.

15.3.5 IF terminal does not respond or does not meet any of criteria above,
THEN **DEPRESS** reset key once and **LOG ON** again.

15.3.6 IF above criteria is not met or specified sequence of events does not occur,
THEN **CONTACT** Information Technology (IT) Department for assistance.

16. ACTIVATING A TURN-ON-CODE

16.1 IF display currently operating in area of screen that desired TOC requires,
THEN **DEPRESS** CANC key.

16.2 WHEN "SELECT FUNC. KEY OR TURN ON CODE..." is displayed followed by an open IDE field,
THEN **ENTER** one of following:

16.2.1 A TOC (i.e., "GROUP" -- activates Group Display Program; program will then prompt user to select a menu option).

16.2.2 A TOC followed by a space and optional text (i.e., "PLOT ARM1" -- activates Real-Time Plot Program and plots group "ARM1" without further user input; note that optional text is recognized by only selected TOCs).

16.2.3 **DEPRESS** one of programmable function keys on right hand keypad or top row of function keys (i.e., blue "GROUP DISP" key -- functions same as first example).

16.3 **REFER** to FKA for function keys that are enabled and their descriptions. Use other options as provided by each program.

16.4 To exit program, **USE** specified exit option (if provided) or **DEPRESS** CANC key.

17. DETERMINING TO WHICH SYSTEM A TERMINAL IS CONNECTED

17.1 PMIS System to which a terminal is connected is indicated by "CONSOLE =..." on bottom line of FKA as follows:

- CONSOLE = PRIMARY - Connected to Primary System operating in Primary Mode.
- CONSOLE = PRIM/BAC - Connected to Primary System operating in Primary/Backup Mode.
- CONSOLE = BACKUP - Connected to Backup System.
- CONSOLE = UNKNOWN - PMIS is in a transition or unknown state.

ATTACHMENT 13 PMIS MET AND RAD DATA POINT INFORMATION (INFORMATION USE)

1. WHEN normal source of meteorological data (PMIS DOSEIN or MET screen) not available or "unhealthy",
   THEN **OBTAIN** data by PMIS Point ID.

2. **USE** Table 11-1 as a guide for selection of PMIS data points.

3. Train "A" is preferred train. However, "A" train and "B" train instruments are all calibrated to same standards, so use of "B" instruments with "A" instruments is acceptable.

4. PMIS Turn-On-Code "VALUE" is used to display single point values and qualities.

5. PMIS Turn-On-Codes "DOSEIN" or "MET" is used to display most meteorological point values and stability classes.

6. IF a significant difference between companion instruments (same parameter and same level) is evident,
   THEN **DELETE** erroneous instrument from PMIS processing per Procedure 2.6.3PMIS to assure accurate dose projections.

7. IF PMIS not available,
   THEN **CONTACT** National Weather Service (NWS) at 402-359-4381 or refer to telephone directory and **OBTAIN** data.

8. IF cannot contact NWS,
   THEN **USE** default MET values.

Table 11-1

PMIS POINT ID	DESCRIPTION
MET001	100M(A) SIGMA THETA
MET002	100M(A) WIND SPEED
MET003	100M(A) WIND DIRECTION
MET004	100M(A) TEMPERATURE
MET005	100-10M(A) DELTA TEMPERATURE
MET006	100M(A) WIND DIR. (15MIN AVE)
MET007	100M(A) WIND SPEED (15MIN AVE)
MET009	60M(A) SIGMA THETA
MET010	60M(A) WIND SPEED
MET011	60M(A) WIND DIRECTION
MET012	60M(A) TEMPERATURE

Table 11-1	
PMIS POINT ID	DESCRIPTION
MET013	100-60M(A) DELTA TEMPERATURE
MET014	60M(A) WIND DIR. (15MIN AVE)
MET015	60M(A) WIND SPEED (15MIN AVE)
MET017	10M(A) SIGMA THETA
MET018	10M(A) WIND SPEED
MET019	10M(A) WIND DIRECTION
MET020	10M(A) TEMPERATURE
MET021	60-10M(A) DELTA TEMPERATURE
MET022	DEW POINT TEMPERATURE
MET023	10M(A) WIND DIR. (15MIN AVE)
MET024	10M(A) WIND SPEED (15MIN AVE)
MET027	PRECIPITATION RATE (LAST 15-MIN)
MET032	100M(A) TEMPERATURE (15MIN AVE)
MET033	100-10M(A) DELT TEMP (15 MIN AV)
MET034	60M(A) TEMPERATURE (15MIN AVE
MET035	100-60M(A) DELT TEMP (15MIN AVE)
MET101	100M(B) SIGMA THETA
MET102	100M(B) WIND SPEED
MET103	100M(B) WIND DIRECTION
MET104	100M(B) TEMPERATURE
MET105	100-10M(B) DELTA TEMPERATURE
MET106	100M(B) WIND DIR. (15MIN AVE)
MET107	100M(B) WIND SPEED (15MIN AVE)
MET109	60M(B) SIGMA THETA
MET110	60M(B) WIND SPEED
MET111	60M(B) WIND DIRECTION
MET112	60M(B) TEMPERATURE
MET113	100-60M(B) DELTA TEMPERATURE
MET114	60M(B) WIND DIR. (15MIN AVE)
MET115	60M(B) WIND SPEED (15MIN AVE)
MET117	10M(B) SIGMA THETA
MET118	10M(B) WIND SPEED
MET119	10M(B) WIND DIRECTION
MET120	10M(B) TEMPERATURE
MET121	60-10M(B) DELTA TEMPERATURE
MET123	10M(B) WIND DIR. (15MIN AVE)
MET124	10M(B) WIND SPEED (15MIN AVE)
MET132	100M(B) TEMPERATURE (15MIN AVE)
MET133	100-10M(B) DELT TEMP (15MIN AV)
MET134	60M(B) TEMPERATURE (15MIN AVE)
MET135	100-60M(B) DELT TEMP (15MIN AVE

Table 11-1	
PMIS POINT ID	DESCRIPTION
MET136	10M(B) TEMPERATURE (15MIN AVE)
MET137	60-10M(B) DELTA TEMP(15MIN AVE)
N8000	RX BLDG EFFLUENT FLOW AVE
N8001	TURB BLDG EFF HI RAD MON AVE
N8002	TURB BLDG EFF NORM RAD MON AVE
N8003	TURB BLDG FLOW AVE
N8004	AOG & RW EFF HI RAD MON AVE
N8005	AOG & RW EFF NORM RAD MON AVE
N8006	RX BLDG EFF RAD MON AVE
N8007	AOG & RW BLDG EFF FLOW AVE
N8010	ERP HI RAD MON AVE
N8011	ERP NORMAL RAD MON AVE
N8012	ERP FLOW AVE
N8013	SGT FLOW TO ERP AVE

ATTACHMENT 14 INFORMATION SHEET (INFORMATION USE)

1. PURPOSE℗[1]

1.1 Perform plume centerline or non-centerline dose assessment including dose projections from Multiple Source Releases, using hand calculated methods when CNS-DOSE computer program is not available.©[6]

1.2 Perform core damage estimate using in-containment high range radiation monitors during post-LOCA inside primary containment.©[1]

2. DISCUSSION

2.1 Manual dose projection methods are used when CNS-DOSE is unavailable.

2.2 Following items are necessary to support performance of hand calculation dose assessment:

- Environs map.
- Scientific calculator.
- Computer terminals.
- Multi Source Release Summation Excel Spreadsheet.

2.3 Where possible, data used is from same source as that used by CNS-DOSE computer program.

2.4 Hand calculations are divided into two attachments.

2.4.1 Attachment 1 is intended to be used by on-shift or Dose Assessment personnel for centerline dose projections.

2.4.2 Attachment 5 is intended for Dose Assessment Personnel in projecting non-centerline values for specific receptor locations.

2.5 Attachments 3 and 4 are intended for centerline, and Attachment 7 for non-centerline, provide EOF Dose Assessment personnel with a means of correlating Field Monitoring Team dose rate readings with hand calculated dose projections.

2.5.1 Such a correlation is necessary to determine if initial Protective Action Recommendations (PARs) were adequate to protect health and safety of public.

2.5.2 When correlating non-centerline dose projections, direct field sample readings are at receptor location requiring no adjustments in order to compare against hand calculated non-centerline dose assessment results.

2.5.3 It is expected hand calculated dose projection results, either centerline or non-centerline, are higher or conservative as compared to direct or adjusted Field Monitor Team readings.

2.5.4 If Field Monitoring Team sample results exceed hand calculated dose projection results, then adjustments to hand calculation is required to correct to field measured conditions.

2.5.5 Control Room is not required to perform this as specified in NUREG 0654, Revision 1, Pages 2 through 9.

2.6 Attachment 8 provides supplemental guidance for determinations of degraded core conditions. The determination of degraded core during hand calculation of dose has a significant impact on contribution from Iodine to Noble Gas Ratio as well as TEDE and CDE conversion factors. Under same radiological release rate, if dose assessment assumes degraded core, resulting CDE dose rate may be as much six decades higher than non-degraded core assessment.

2.7 Attachment 9 provides guidance to estimate core damage in cases where high range in-containment radiation monitors are exposed to coolant or steam (i.e., only for primary containment LOCA situations).

2.7.1 It is an indication of inventory of airborne fission products (i.e., noble gases, a fraction of halogens, and a much smaller fraction of particulates) released from fuel to containment (refer to NEDO-22215, Pages 1 and 2; NEDC 02-009).©[1]

2.7.2 For other accident sequences, Post-Accident Sampling System (PASS) may be used, as required, to obtain Reactor Coolant System (RCS) sample.©[7]

2.8 Procedure provides instructions for obtaining MET data from alternate sources if primary sources unavailable.

2.8.1 General order of preference is PMIS, National Weather Service, and then historically determined default values.

2.9 If user is not familiar with use of PMIS, Attachment 12 provides an overview and instructions on access and selected use.

2.10 Attachment 13 provides a list of PMIS MET and RAD data points that can be used to perform hand calculated dose assessment.

3. RECORDS

3.1 Those portions of completed worksheets, for actual events, from attachments listed below shall be forwarded to EP Manager within five working days of their completion (quality record upon completion):

- Attachment 2.
- Attachment 4.
- Attachment 6.
- Attachment 7.
- Attachment 9.

4. REFERENCES

4.1 TECHNICAL SPECIFICATIONS

4.1.1 Technical Specification Bases B.3.3.6.1 (2.d), Main Steam Line Radiation - High.

4.2 CODES AND STANDARDS

4.2.1 EPA 400-R-92-001, May 1992, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents.

4.2.2 Health Physics Journal, November 1981, Noble Gas Dose Rate Conversion Factors.

4.2.3 ICRP 59, Working Breathing Rate.

4.2.4 NRC Regulatory Guide 1.23, Revision 1, March 2007, Meteorological Programs for Nuclear Power Plants.

4.2.5 NRC Regulatory Guide 1.109, Revision 1, October 1977, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR50, Appendix I, Iodine Inhalation Dose Factors.

4.2.6 NRC Regulatory Guide 1.111, July 1977, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors.

4.2.7 NRC Regulatory Guide 1.145, August 1979, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.

4.3 DRAWINGS (MAPS)

4.3.1 Cooper Nuclear Station 50 Mile Emergency Planning Zone, 50 Mile Radius.

4.3.2 NPPD Drawing CNS-MI-03, Preselected Radiological Sampling and Monitoring Points in the Vicinity of Cooper Nuclear Station, 10 Mile Radius.

4.3.3 NPPD Drawing CNS-MI-102, Atmospheric Dispersion Model (EPM2) Special Receptor Points, 10 Mile Radius.

4.3.4 NPPD Drawing 2.2 (P3-A-45), Cooper Nuclear Station Site and Property Boundary, 1 Mile Radius.

4.4 PROCEDURES

4.4.1 Computer System Operating Procedure 2.6.3PMIS, PMIS Computer System Operation and Outage Recovery.

4.4.2 Emergency Plan Implementing Procedure 5.7.1, Emergency Classification.

4.4.3 Emergency Plan Implementing Procedure 5.7.16, Release Rate Determination.

4.4.4 Emergency Plan Implementing Procedure 5.7.17, CNS-DOSE Assessment.

4.4.5 Emergency Plan Implementing Procedure 5.7.20, Protective Action Recommendations.

4.5 MISCELLANEOUS

4.5.1 Engineering Evaluation EE 02-056, Elimination of Meteorological Instrumentation System Strip Chart Recorder References.

4.5.2 General Electric Corporation, NEDO-22215, Procedures for the Determination of the Extent of Core Damage Under Accident Conditions.©[1]

4.5.3 NEDC 99-034, Control Room, EAB, and LPZ Doses Following a CRDA.

4.5.4 NEDC 02-004, Estimation of Steam Jet Air Ejector Radiation Monitor, RMP-RM-150A (B), Readings Following a 1% Fuel Clad Release (Degraded Core) in Reactor Coolant System.

4.5.5 NEDC 02-009, Estimation of Primary Containment High Range Monitor, RMA-RM-40A (B), Readings Following 1% Clad Failure in RCS Under Non-LOCA Conditions.

4.5.6 NEDC 02-20, Estimation of Reactor Coolant System Dose Equivalent I-131 Concentration Following a 1% Fuel Clad Failure (Degraded Core) Under Non-LOCA Conditions.

4.5.7 NEDO-31400, Safety Evaluation for Eliminating BWR MSIV Closure Function and Scram Function for MSL Rad Monitors.

4.5.8 NEI 99-01, Revision 5, SER TAC Number ME0840.

4.5.9 SAIC Document Number 502-85500107-72, PMIS Operator's Manual.

4.5.10 ℗[1] TIP Action Plan 5.2.2.1, Revision 1, Action 1. Major revision 6/28/02 to clarify Control Room tasks. Affects entire procedure (Attachment 8, Step 1 flagged).

4.6 NRC Commitments

4.6.1 ©[1] NRC Commitment NLS91000613-01. NPPD Response to IR 91-12, Include a method for estimating Core Damage using In-Containment Radiation Monitors. Commitment affects Section 2, Table 1, Line 3; Attachment 9; and Attachment 14, Steps 1.2, 2.7.1, and 4.5.2.

4.6.2 ©[2] NRC Commitment NLS91000613-05. NPPD Response to IR 91-12, Assure the correct input for the Standby Gas Treatment System is selected. Commitment affects Attachment 1, NOTE prior to Step 5 and Step 5.1; and Attachment 5, NOTE prior to Step 5 and Step 5.1.

4.6.3 ©[3] NRC Commitment NLS91000613-06. NPPD Response to IR 91-12, Facilitate correlation of Field Monitoring Team data with dose projections and facilitate verification of protective action recommendations. Commitment affects Attachment 1, Step 28; Attachment 5, Step 27; and Attachments 3, 4, and 7.

4.6.4 ©[4] NRC Commitment NSD921204-07. IR 92-14 Corrective Actions, Cladding Damage criterion clarified. Commitment affects Attachment 1, Step 7.1; Attachment 5, Step 7.1; and Attachment 8, Step 1.

4.6.5 ©[5] NRC Commitment NLS8900460-03 and 04. NPPD Response to IR 89-35, Commit 3 was to stress importance of selecting correct setting for "core degraded" input to dose assessment computer models and Commit 4 was to require coordination between Engineering Operations, and Health Physics personnel in determining "core degraded" status. Commitment affects Attachment 1, Step 7.2; Attachment 5, Step 7.2; and Attachment 8, NOTE prior to Step 1.

4.6.6 ©[6] NRC Commitment 811217-01-07. NPPD Response to IR 81-13, Develop a Functional Procedure for Dose Assessment. Commitment affects Attachment 14, Step 1.1.

4.6.7 ©[7] NRC Commitment 811217-01-24. NPPD Response to IR 81-13, Proper utilization of new post-accident sampling system will be addressed in appropriate EPIPs. Commitment affects Attachment 9, NOTE prior to Step 1 and Step 1; and Attachment 14, Step 2.7.2.

4.6.8 ©[9] NRC Commitment NLS2014035-02. Maintain capability for classifying fuel damage events at the Alert level threshold for CNS at radioactivity levels of 300 μCi/cc dose equivalent iodine. Commitment affects Attachment 8, Step 1.1.

4.6.9 ©[10] NRC Commitment 830513-02-05. NPPD Follow-up Response to IR 81-13, Modify hand calculation technique to calculate concentrations from ground-level type releases for each stability class category. Commitment affects Attachment 1, Step 17; Attachment 2, Table 2-2, Block 14; and Attachment 5, NOTE prior to Step 16.2 and Step 16.2.

Enclosure 5

Emergency Plan Implementing Procedure 5.7.17, Revision 45

COOPER NUCLEAR STATION

Operations Manual

Emergency Preparedness

EMERGENCY PLAN IMPLEMENTING PROCEDURE

5.7.17

CNS-DOSE ASSESSMENT

Level of Use:	**MULTIPLE**
Quality:	**QAPD RELATED**
Effective Date:	**3/28/16**
Approval Authority:	**ITR-RDM**
Procedure Owner:	**EP ON-SITE COORDINATOR**

TABLE OF CONTENTS

1. ENTRY CONDITIONS (INFORMATION USE)

1.1 Actual or projected radiological release above NOUE release rate limits has occurred requiring dose assessment per Emergency Director with CNS-DOSE available.

1.1.1 IF CNS-DOSE unavailable,
THEN **GO TO** EPIP 5.7.17.1 to perform hand calculation of projected dose.

2. INSTRUCTIONS (INFORMATION USE)

2.1 **PERFORM** CNS-DOSE per PMIS on-line instruction.

2.1.1 **USE** Attachment 1, CNS-DOSE Guide, as necessary.

2.2 **PERFORM** Attachment 2 to compare CNS-DOSE to field monitoring sample results.

2.3 IF any of following occur,
THEN **RE-PERFORM** CNS-DOSE assessment:

- Significant meteorological changes in wind speed, wind direction, and precipitation.
- Every hour for effective age 0-10 hours.
- Every 10 hours for effective age 10-100 hours.
- Conditions reach critical parameter per Emergency Director (ED)/Radiological Assessment Supervisor (RAS).

ATTACHMENT 1 CNS-DOSE GUIDE (INFORMATION USE)©8

NOTE – Attachment 7, Step 15, provides assistance for use of PMIS login information and turn-on code activation.

1. **ENTER** CNS-DOSE program:

   1.1 At PMIS IDT, **ENTER** turn-on code DOSE or **PRESS** DOSE key on terminal logged into either Primary or Backup PMIS System.

   1.2 At PC that runs CNS-DOSE, **SELECT** CNS-DOSE program from menu.

2. **ENTER** Y or N at prompt, "Is this a REAL EMERGENCY? (Yes/No)".

3. On MET/RAD INPUTS screen, **PERFORM** following:

   3.1 **HIGHLIGHT** and **EDIT** fields, as necessary.

NOTE – Name for Release Source must be listed in "Release Source" column before CNS-DOSE will allow release rate or duration values to be entered.

   3.2 **ENSURE** name of release source to be analyzed is entered in Release Source column.

      3.2.1 IF list of Release Sources appropriate, THEN **GO TO** Step 3.3.

      3.2.2 IF Release Source needs to be added, THEN **PERFORM** following:

         3.2.2.1 **HIGHLIGHT** "(other)" in Release Source column.

         3.2.2.2 **ENTER** name of other release source (e.g., sea land, tank trailer, etc.).

         3.2.2.3 IF Release Source "(other)" needs to be deleted, THEN **USE** F6 function key.

   3.3 **VERIFY** release rates available and health GOOD in "Rate (µCi/sec)" column for each release source listed.

      3.3.1 IF "Rate (µCi/sec)" values appropriate and health GOOD, THEN **GO TO** Step 3.4.

3.3.2 IF release rate values unavailable or health BAD,
THEN **PERFORM** following:

3.3.2.1 **HIGHLIGHT** health BAD or blank "Rate (μCi/sec)" value for selected release source.

3.3.2.2 **ENTER** release rate in μCi/sec for release source as determined per EPIP 5.7.16.

3.3.2.3 **VERIFY** health code changed from GOOD to OSUB to denote substituted value.

NOTE – Four hour release durations is default value used in CNS-DOSE to expedite dose assessment unless directed otherwise.

3.4 **VERIFY** value "4.0" entered in "Duration (hrs)" column unless other duration desired.

3.4.1 IF "Duration (hrs)" value appropriate,
THEN **GO TO** Step 3.5.

3.4.2 IF release duration other than 4.0 hours desired,
THEN **PERFORM** following:

3.4.2.1 **ESTIMATE** duration of release in hours.

3.4.2.2 **CONSULT** with ED or Operations.

3.4.2.3 IF no release in-progress or duration unknown,
THEN **USE** default 4 hours.

3.4.2.4 **HIGHLIGHT** duration in any release source.

3.4.2.5 **ENTER** release duration in units of hours and quarter hours.

NOTE – "AOG & RW EFF NORM RAD MON AVE" and "(other)" release source entries for "Rel thru Rx Bldg" are blank and cannot be edited as CNS-DOSE assumes Reactor Bldg hold-up volume not credited in release pathway.

3.5 IF value in "Rel thru Rx Bldg" column for each of following Release Sources appropriate for plant conditions:

- ERP NORMAL RAD MON AVE.
- RX BLDG EFF RAD MON AVE.
- TURB BLDG EFF NORM RAD MON AVE.

THEN **GO TO** Step 3.7.

3.5.1 IF "Rel thru Rx Bldg" value requires change, THEN **PERFORM** following:

3.5.1.1 **DETERMINE** if release pathway is into Reactor Building (credit Rx BLDG for plate-out):

a. **CONSULT** with RCM, ED, or Operations, as available.

3.5.2 Use Table 1 to **SELECT** appropriate entry for release pathway based on release path and plant conditions.

Table 1

RELEASE SOURCE	RELEASE PATH	ENTER
ERP	Reactor Bldg through SGT without Hard Pipe Vent	YES
ERP	Primary Containment through Hard Pipe Vent	NO
ERP	Gland Seal Steam, Main Condenser through Air Removal, or Off-Gas	NO
Reactor Bldg	Reactor Bldg HVAC or direct ground level from blowout panel or other opening	YES
Turbine Bldg	TB HVAC or direct ground level from blowout panel or other opening	NO
Turbine Bldg	TB HVAC plus Reactor Bldg via TB-Reactor Bldg blowout panel	YES
RW/ARW Bldg	RW/ARW HVAC or other direct ground level opening	NO

3.5.3 **CHANGE** value in "Rel thru Rx Bldg" column as follows:

3.5.3.1 **HIGHLIGHT** value in "Rel thru Rx Bldg" column in one of Release Source entries listed in Step 3.5.

3.5.3.2 **ENTER** Y or N.

3.6 IF value for each of following appropriate and health GOOD for elevated MET data:

- 100M WIND DIR. (15MIN AVE) (deg).
- 100M WIND SPEED (15MIN AVE) (mph).
- Stability-Class (A-G).

THEN **GO TO** Step 3.7.

NOTE – MET Tower instrumentation from different trains may be used for each elevation or parameter when performing dose assessment with "A" being preferred train. If 100 meter MET tower is out of service or unavailable, PMIS will default to next highest tower if elevated release. Default is 60M.

3.6.1 IF value not appropriate or health BAD for elevated MET data, THEN **PERFORM** following:

3.6.1.1 **USE** data in following preferred order for elevated wind speed for 15 minute average:

a. PMIS 100 meter level.

b. PMIS 60 meter level.

c. **CONTACT** NWS at 402-359-4381 or refer to telephone directory and **OBTAIN** average wind speed estimate for 100 meter elevation.

d. Historical default wind speed value of 13 mph.

3.6.1.2 **ENTER** wind speed value.

3.6.1.3 **VERIFY** desired wind speed value entered.

3.6.1.4 **USE** data in following preferred order for elevated wind direction (degree from) for 15 minute average:

a. PMIS 100 meter level.

b. PMIS 60 meter level.

c. **CONTACT** NWS at 402-359-4381 or refer to telephone directory and **OBTAIN** CNS estimate of wind direction (degree from) at 100 meter elevation.

3.6.1.5 **ENTER** elevated wind direction.

3.6.1.6 **VERIFY** desired wind direction entered.

3.6.1.7 **USE** data in following preferred order for atmospheric stability class ("A" through "G"):

a. **DIRECT** PMIS stability class readout.

b. **REFER** to EPIP 5.7.17.1 to calculate stability class.

3.6.1.8 **ENTER** stability class.

3.6.1.9 **VERIFY** desired stability class entered.

NOTE – Multiple release paths may exist concurrent with SGT running and in flow path to ERP, including unfiltered radiological release pathway requiring entering N for No for entry on "Released Through SGT?".©2

3.7 **VERIFY** entry for "Released Through SGT?" appropriate for all release sources and pathways based on plant conditions.

3.7.1 **CONSULT** with following:©2

- Operations.
- RCM or Dose Assessor.
- Engineering, if available.

3.7.2 IF certain of "Released Through SGT?" entry,
THEN **GO TO** Step 3.9.

3.7.3 IF either of following conditions met:
- Uncertain of "Released Through SGT?" entry.
- Multiple release paths exist.

THEN use Table 2 to determine most appropriate answer.

Table 2

RELEASE SOURCE	RELEASE FILTERED THROUGH SGT	ENTER
ERP	Yes: Rx Bldg	YES
ERP	No: Hard Pipe Vent	NO
ERP	No: Gland Seal Steam, Main Condenser through Air Removal, and Off-Gas	NO
Reactor Bldg	No: HVAC or direct ground level from blowout panel or other opening	NO
Turbine Bldg	No: TB HVAC or direct ground level from blowout panel or other opening	NO
Turbine Bldg	No: TB HVAC plus Reactor Bldg via TB-Reactor Bldg blowout panel	NO
RW/ARW Bldg	No: RW/ARW HVAC or other direct ground level opening	NO

3.7.4 IF entry for "Released Through SGT?" requires change, THEN **PERFORM** following:

3.7.4.1 **HIGHLIGHT** "Released Through SGT?" current value.

3.7.4.2 **ENTER** Y or N.

3.7.4.3 **VERIFY** desired "Released Through SGT?" value entered.

3.8 IF value for each of following appropriate and health GOOD for ground level release MET data:
- 10M WIND DIR. (15MIN AVE) (deg).
- 10M WIND SPEED (15MIN AVE) (mph).
- Stability-Class (A-G).

THEN **GO TO** Step 3.9.

NOTE – MET Tower instrumentation from different trains may be used for each elevation or parameter when performing dose assessment with "A" being preferred train. If 10 meter MET tower is out of service or unavailable, PMIS will default to next highest tower if ground level release. Default is 60M.

3.8.1 IF value not appropriate or health BAD for ground level release MET data,
THEN **PERFORM** following:

3.8.1.1 **USE** data in following preferred order for ground level release wind speed 15 minute average:

a. PMIS 10 meter level.

b. PMIS 60 meter level.

c. **CONTACT** NWS at 402-359-4381 or refer to telephone directory and **OBTAIN** average wind speed estimate for 10 meter elevation.

d. Historical default wind speed value of 8 mph.

3.8.1.2 **ENTER** wind speed value.

3.8.1.3 **VERIFY** desired wind speed value entered.

3.8.1.4 **USE** data in following preferred order for ground level release wind direction (degree from) for 15 minute average:

a. PMIS 10 meter level.

b. PMIS 60 meter level.

c. **CONTACT** NWS at 402-359-4381 or refer to telephone directory and **OBTAIN** CNS estimate of wind direction (degree from) at 10 meter elevation.

3.8.1.5 **ENTER** wind direction value.

3.8.1.6 **VERIFY** desired wind direction entered.

3.8.1.7 **USE** data in following preferred order for atmospheric stability class ("A" through "G"):

a. **DIRECT** PMIS stability class readout.

b. **REFER** to EPIP 5.7.17.1 to hand-calculate stability class.

c. **ENTER** stability class.

3.8.1.8 **VERIFY** desired stability class entered.

3.9 **VERIFY** entry for "Degraded Core? (Y/N)" appropriate based on plant conditions.

3.9.1 **REFER** to Attachment 5 for guidance on degraded core.©[4]

3.9.2 **CONSULT** with following to coordinate answer:©[5]

- Operations.
- RP/Chemistry, if available.
- Engineering, if available.

3.9.3 IF "Degraded Core? (Y/N)" appropriate,
THEN **GO TO** Step 3.10.

3.9.4 IF entry for "Degraded Core? (Y/N)" requires change,
THEN **ENTER** Y or N.

3.9.5 **VERIFY** desired "Degraded Core? (Y/N)" value entered.

NOTE – Reactor shutdown under all conditions met by any of following:
- All rods fully inserted.
- Reactor scram and power below 3%.
- Cold shutdown boron weight injected per EOPs.

3.10 **OBTAIN** Time of Shutdown under all conditions from Operations.

3.10.1 IF Reactor not shutdown under all conditions or Time since Shutdown unknown,
THEN **ENTER** zero.

3.10.2 **RECORD** Time of Shutdown: ________.

3.10.3 **ENTER** Time Since Shutdown under all conditions (hrs).

3.11 **OBTAIN** Dose Assessment results.

3.11.1 IF IDT,
THEN **DEPRESS** Function Key F3 or **ENTER** "R" to process.

3.11.2 IF PC,
THEN **ENTER** "R" to process.

4. **CONSULT** with ED or Operations to respond to following questions:

4.1 **ENTER** Y or N to following question: "Declare a General Emergency based on PLANT CONDITIONS? (N/Y)".

4.1.1 IF answer "YES",
THEN **GO TO** Step 4.2 for question regarding Rapidly Progressing Severe Accident (RPSA).

4.1.2 IF answer "NO",
THEN **GO TO** Step 5 to print Dose Assessment results.

NOTE – Rapidly Progressing Severe Accident generates default PARs that override PARs based on dose assessment.

4.2 **USE** Figure 1-1 to respond Y or N to following question: "Is this a Rapidly Progressing Severe Accident? (N/Y)."

4.2.1 **CONSULT** with ED or Operations.

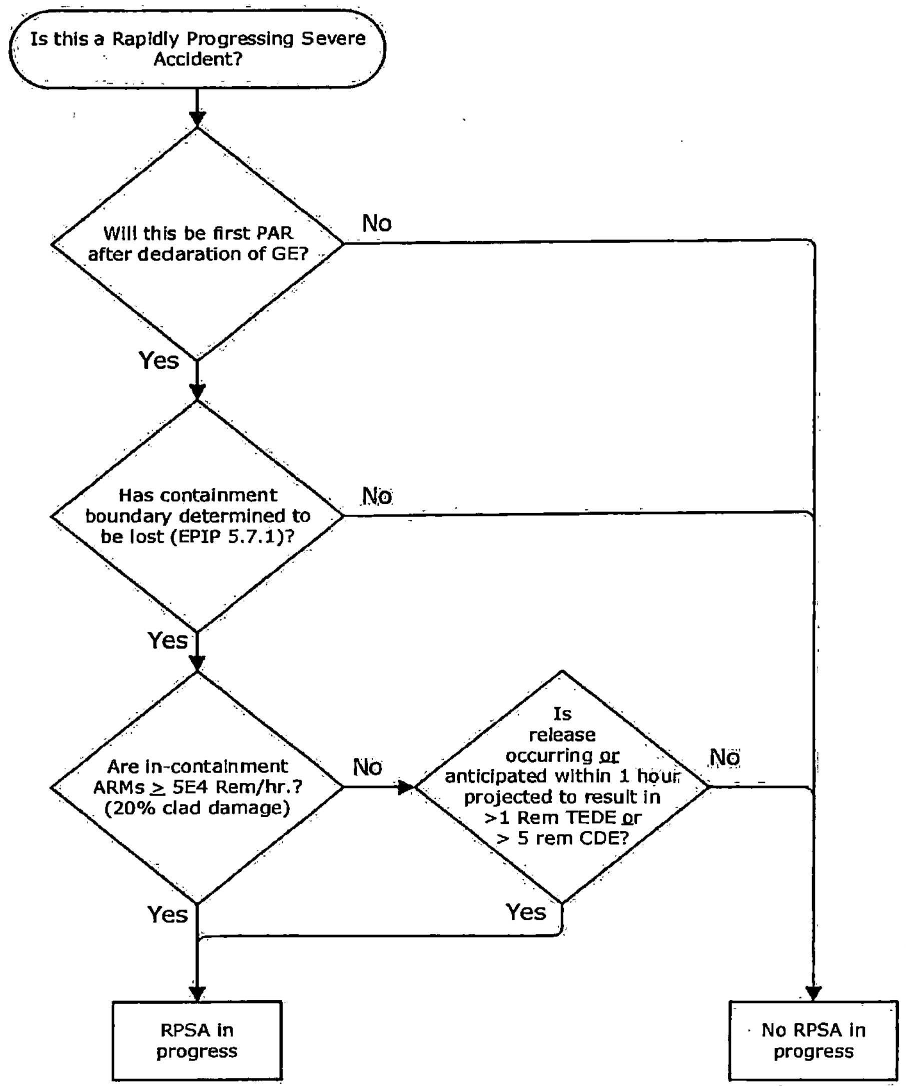


Figure 1-1 – Rapidly Progressing Severe Accident Determination

4.2.2 IF answer "YES",
THEN **GO TO** Step 5 to print Dose Assessment results.

4.2.3 IF answer "NO",
THEN **ENTER** "N" and **PROCEED** to next step.

4.3 **ENTER** Y or N to following question: "Evacuation PARS Previously Recommended? (Y/N)."

4.3.1 IF answer "YES",
THEN **GO TO** Step 5 to print Dose Assessment results.

4.4 **ENTER** Y or N to following question: "Are there Known Impediments to Evacuation? (Y/N)."

4.4.1 IF answer "YES",
THEN **GO TO** Step 5 to print Dose Assessment results.

4.5 **ENTER** Y or N to following question: "Is this a Controlled-Release of less-than 1 hour? (Y/N)."

5. **SELECT** Function-Key (F5) to print Dose Assessment results.

6. IF desired to prepare Look-Ahead values on Dose Assessment Results screen to project escalated release rates from specific release source,
THEN **PERFORM** following:

6.1 **SELECT** "Look-Ahead" option.

6.2 **CHOOSE** release source to look ahead at release rates for escalation.

7. IF desired to change data,
THEN **ENTER** "M" for "MET/RAD-IN" in Options entry box and **PERFORM** following:

7.1 **OBTAIN** new data by entering "NE" in Options entry box for new sample and make additional changes, as necessary.

7.2 **REPEAT** Steps 3 through 5 to produce Dose Assessment Report.

8. WHEN Field Monitoring Teams report sample results to EOF,
THEN **GO TO** Attachment 2 to correlate field sample results with CNS-DOSE projections.

9. IF off-site dose exceeds 1 rem TEDE or 5 rem CDE at or beyond boundary of Owner Controlled Area (OCA),
THEN **NOTIFY** ED or RCM.

ATTACHMENT 2 CORRELATING FIELD MONITORING SAMPLE RESULTS WITH CNS-DOSE PROJECTIONS (REFERENCE USE)©3

1. **PERFORM** flowchart in Figure 1 to compare CNS-DOSE with Field Monitoring Team sample results.

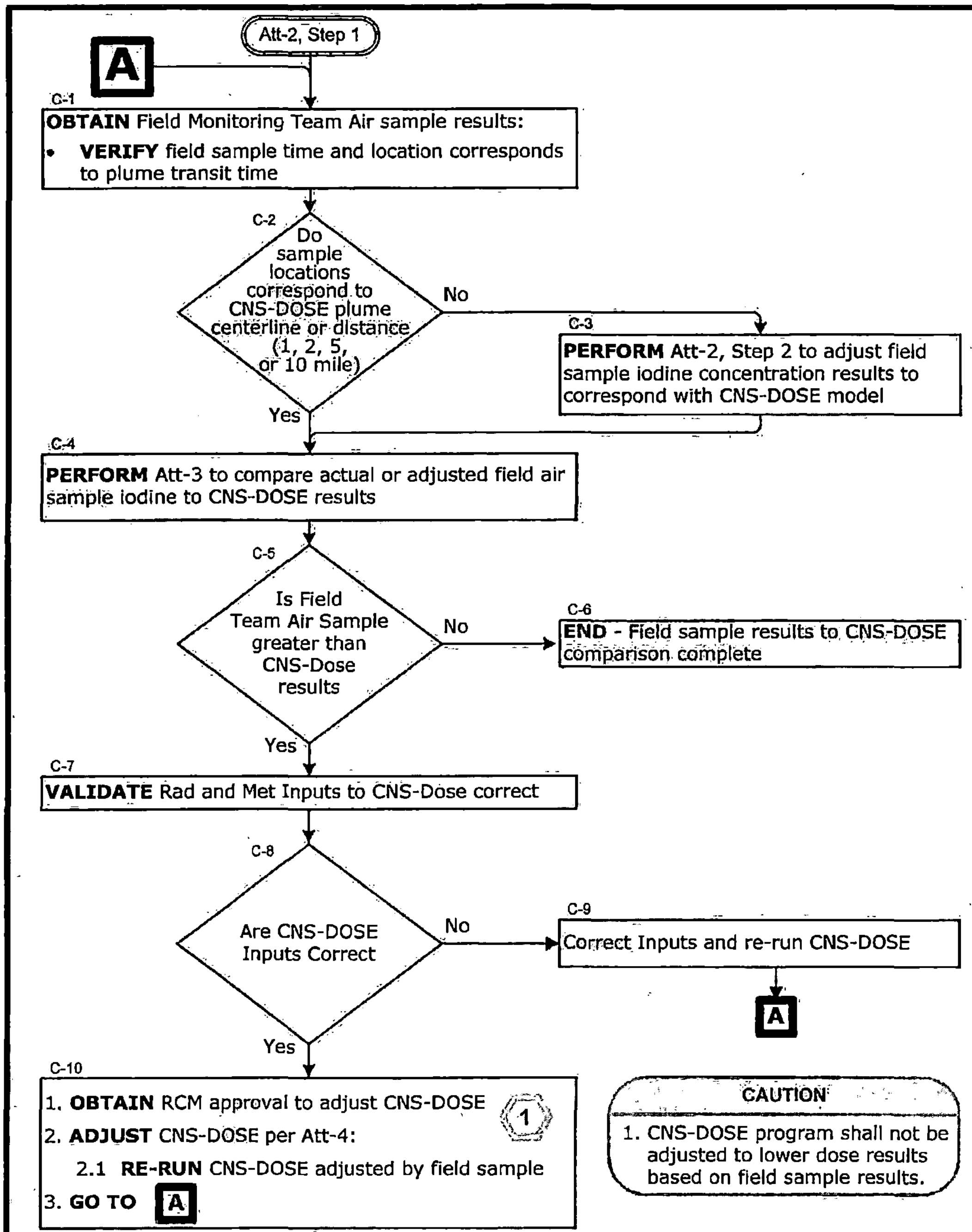


Figure 1 – Field Sample Comparison to CNS-DOSE Decision Flowchart

2. **PERFORM** adjustments to Field Monitoring Team air sample results to correspond to plume centerline or distance of 1, 2, 5, or 10 miles, as follows:

   2.1 IF Field Monitoring Team air sample results are aligned to centerline of wind direction or were taken at highest radiological point during traverse of plume,
   THEN **GO TO** Step 2.3.

   2.2 **PERFORM** following to determine non-centerline adjustment ratio for field sample.

   2.2.1 **RECORD** field sample iodine reading in μCi/cc in Table 1, Column (1).

   2.2.2 **SELECT** isopleth based on atmospheric stability class.

   NOTE – Isopleth overlays are designed to scale to only 10 mile EPZ map with 1 mile to 1 inch scale. Additional extrapolation required if applying to 50 mile EPZ map.

   2.2.3 **PLACE** selected isopleth onto 10 mile EPZ map with isopleth asterisk zero directly over zero point on map representing plant.

   2.2.3.1 **VERIFY** map and isopleth align to correct scale.

   2.2.4 **OBTAIN** downwind direction during period associated with field sample.

   2.2.5 **ALIGN** isopleth centerline on 10 mile EPZ map according to downwind direction centerline.

   2.2.6 **LOCATE** field sample location on 10 mile EPZ map.

   2.2.7 **RECORD** alphabetic designation of isopleth segment of plume in Table 1, Column (2), that field sample is located in.

NOTE – Interpolation between isopleth segment X/Q values may be used to increase accuracy when sample location lies between isopleth segment lines.

2.2.8 **USE** X/Q VALUES table, ISOPLETHS column, to determine X/Q value associated with field sample isopleth segment recorded in Table 1, Column (2).

2.2.8.1 **RECORD** X/Q value associated with isopleth segment containing field sample location in Table 1, Column (3).

2.2.9 At same distance as field sample location from release point, **DETERMINE** alphabetic designation of isopleth segment of plume centerline.

2.2.9.1 **RECORD** alphabetic designation of isopleth segment of plume centerline in Table 1, Column (4).

2.2.10 **USE** X/Q VALUES table, ISOPLETHS column, to determine X/Q value associated with isopleth segment of plume centerline recorded in Column (4).

2.2.10.1 **RECORD** X/Q value associated with isopleth segment of plume centerline in Table 1, Column (5).

NOTE – X/Q value associated with each isopleth segment away from centerline is factor of 10 lower than centerline. Therefore, to adjust off-centerline sample to centerline, sample must be multiplied by factors of 10 based on number of segments away from centerline.

2.2.11 **DETERMINE** plume centerline adjustment ratio between field sample X/Q value and plume centerline X/Q value.

2.2.11.1 **DIVIDE** X/Q value associated with isopleth segment of plume centerline in Table 1, Column (5), by X/Q value associated with isopleth segment containing field sample location in Column (3).

2.2.11.2 **RECORD** plume centerline adjustment ratio in Table 1, Column 6.

2.2.12 **MULTIPLY** Field Sample reading Table 1, Column (1), by plume centerline adjustment ratio Column (6) to determine non-centerline adjusted field sample.

2.2.12.1 **RECORD** non-centerline adjusted field sample results in Table 1, Column 7.

Table 1

(1) Field Sample Iodine Concentration (μCi/cc)	(2) Isopleth Segment at Sample Location	(3) X/Q Value for Isopleth Sector at Sample Location	(4) Isopleth Segment at Plume Centerline at Sample Distance	(5) X/Q Value for Isopleth Sector at Plume Centerline	(6) Plume Centerline Adjustment Ratio (5) ÷ (3) (> 1.0)	(7) Non-Centerline Adjusted Field Sample Iodine Concentration (μCi/cc) (1) × (6)
		______		______	______	
		______		______	______	

2.2.13 IF sample distance associated with adjusted non-centerline field sample iodine concentration results in Table 1, Column (7), is aligned to CNS-DOSE distances of 1, 2, 5, or 10 miles, THEN **PERFORM** Attachment 3 using adjusted field sample iodine concentration results in Table 1, Column (7), to compare against CNS-DOSE iodine concentration at same distance.

2.2.14 IF sample distance associated with adjusted non-centerline field sample iodine concentration results in Table 1, Column (7), is not aligned to CNS-DOSE distances of 1, 2, 5, or 10 miles, THEN **PERFORM** Step 2.3 to adjust non-centerline field sample iodine concentration to CNS-DOSE distances.

NOTE – Adjusting centerline to 1 or 2 miles from 1.5 mile sample point centerline will not change dose assessment results, either is acceptable.

2.3 **PERFORM** distance adjustment of either field sample iodine concentration or non-centerline adjusted field sample iodine concentration recorded in Table 1, Column (7).

2.3.1 **RECORD** one of following in Table 2, Column (1):

- Direct field sample iodine concentration whose location was on plume centerline.
- Adjusted non-centerline field sample iodine concentration results from Table 1, Column (7).

2.3.2 **RECORD** actual field sample distance from release source in miles in Table 2, Column (2).

NOTE – If between mile marker designations, interpolate between the 2 mile designators.

2.3.3 At Isopleth, **USE** X/Q VALUES table, CENTERLINE column, to determine X/Q value associated with actual field sample distance from release source in miles recorded in Table 2, Column (2).

2.3.3.1 **RECORD** X/Q value in Table 2, Column (3), associated with actual field sample distance recorded in Column (2).

2.3.4 **RECORD** standard CNS-DOSE distance of 1, 2, 5, or 10 miles associated with CNS-DOSE projection iodine results being evaluated in Table 2, Column (4).

2.3.5 At Isopleth, **USE** X/Q VALUES table, CENTERLINE column, to determine X/Q value associated with CNS-DOSE distance recorded in Table 2, Column (4).

2.3.5.1 **RECORD** X/Q value in Table 2, Column (5), associated with CNS-DOSE distance recorded in Column (4).

2.3.6 **DIVIDE** X/Q value associated with CNS-DOSE distance in Table 2, Column (5), by X/Q value associated with actual field sample distance in Table 2, Column (3), to determine centerline distance adjustment ratio.

2.3.6.1 **RECORD** Centerline Distance Adjustment Ratio in Table 2, Column (6).

2.3.7 **MULTIPLY** field sample or adjusted field sample iodine concentration recorded in Table 2, Column 1, by Centerline Distance Adjustment Ratio in Table 2, Column (6).

2.3.7.1 **RECORD** Adjusted Field Sample Iodine Concentration (μCi/cc) in Table 2, Column (7), and Attachment 3, Column (4).

Table 2

(1) Field Sample or Non-Centerline Adjusted Field Sample [Table 1, Column (7)] (μCi/cc)	(2) Actual Sample Distance on Plume Centerline (1–20 miles)	(3) Centerline Column X/Q Value associated with Actual Field Sample Distance	(4) CNS-DOSE Distance (1, 2, 5, or 10)	(5) Centerline Column X/Q Value Associated with CNS-DOSE Distance	(6) Centerline Distance Adjustment Ratio (5) ÷ (3)	(7) Adjusted Field Sample Iodine Concentration (μCi/cc) (1) × (6)

2.3.8 **GO TO** Attachment 3.

ATTACHMENT 3 CNS-DOSE CORRECTION DETERMINATION USING FIELD SAMPLE RESULTS (REFERENCE USE)©3

1. **COMPLETE** Field Monitoring Team sample information in Table 1, as follows:
   - **CIRCLE** appropriate field sample downwind distance in Column (1).
   - **RECORD** field sample downwind sector in Column (2).
   - **RECORD** field sample time in military format (0000 hrs) in Column (3).
   - **RECORD** Iodine Concentration in Column (4) from field sample or adjusted field sample per Section 2.
2. **RECORD** CNS-DOSE Iodine Concentration in µCi/cc in Table 1, Column (5), for same downwind distance as field sample location selected in Column (1).
3. **COMPARE** Iodine Concentrations of Table 1, Column (4) to Column (5).
4. IF Table 1, Column (4), field sample results greater than Column (5) CNS-DOSE results,
   THEN **ENTER** YES in Column (6).
5. IF Table 1, Column (4), field sample results less than Column (5) CNS-DOSE results,
   THEN **ENTER** NO in Column (6).

Table 1

(1) Field Sample Location Miles Downwind (Circle One)	(2) Field Sample Location Downwind Sector	(3) Field Sample Time (Military Time format, 0000 hrs)	(4) Adjusted Field Sample Iodine Concentration (µCi/cc)	(5) CNS-DOSE Iodine Concentration (µCi/cc)	(6) Is Column (4) Greater than Column (5)? YES or NO
1 2 5 10					
1 2 5 10					

Name/Time/Date: ______________ / ______ / ______

6. **RETURN** to Flowchart Step C-5.

ATTACHMENT 4 ADJUSTING CNS-DOSE PROJECTIONS BASED ON FIELD SAMPLE RESULTS (INFORMATION USE)©[3]

NOTE – Errors in dose assessment projection may occur if CNS-DOSE is adjusted using field sample data that does not represent release conditions present at time CNS-DOSE performed dose assessment. This is especially a concern when adjusting CNS-DOSE based on field sample results during short duration or puff type releases.

1. RCM **REVIEW** Attachment 3 and **AUTHORIZE** adjusting CNS-DOSE using field sample results.

2. On MET/RAD-INPUTS screen, RAS **PERFORM** following:

   2.1 **SELECT** "Field Sample" Option.

NOTE – CNS-DOSE generates dose projection reports based on last field sample information entered overriding any previous field sample data.

   2.2 **SELECT** one of four options of distance from CNS in miles "(1/2/5/10 miles)" based on downwind distance of field sample selected per Attachment 3, Table 1, Column (1), at prompt, and **PRESS** Enter.

   2.3 **SELECT** downwind sector based on field sample location from listed sectors based on MET data used in last Dose Assessment Report and **PRESS** Enter.

NOTE – Although CNS-DOSE will indicate "saved" if field sample value lower than CNS-DOSE program results are entered, CNS-DOSE will not apply this lowered value to re-calculate dose or display dose results that are reduced.

   2.4 **ENTER** Field Sample gross iodine concentration (in μCi/cc) from Attachment 3, Table 1, Column (4).

3. **OBSERVE** CNS-DOSE displays all field information that was added and states it has been saved.

   3.1 **VERIFY** field sample information correct.

4. **PERFORM** Attachment 1, Steps 3.11 through 5, to produce adjusted CNS-DOSE results report.

5. **COMPARE** new PARs to any PARs previously transmitted to off-site authorities.

NOTE – When NEw Sample option is selected at CNS-DOSE MET/RAD INPUTS screen, CNS-DOSE deletes all field sample adjustment information previously entered, generating dose assessment projections based on fresh snapshot of radiological and meteorological conditions.

5.1 IF field sample location was not aligned with wind direction used by CNS-DOSE,
THEN **PERFORM** following:

5.1.1 **DETERMINE** sector sample location is in.

5.1.2 IF PAGs in sector have been exceeded,
THEN **ENSURE** any PARs issued include sample location sector, and those sectors between sample location sector and previously identified sectors based on wind direction at CNS.

ATTACHMENT 5 DEGRADED CORE DETERMINATION GUIDE (INFORMATION USE)©[4]

NOTE – Response to Degraded Core question has a significant impact on Iodine to Noble Gas ratio and resultant dose projection calculations.©[5]

1. Core is considered Degraded if any of following conditions are met or if they were met and have subsequently dropped below threshold:

   1.1 Primary Coolant Activity greater than 300 µCi/gm Dose Equivalent I-131.©[9]

   1.2 DW Rad Monitor reading greater than 2500 Rem/hr.

   1.3 Primary Containment flooding required due to any of following:

   1.3.1 RPV water level cannot be restored and maintained greater than -183 inches.

   1.3.2 RPV water level cannot be restored and maintained greater than or equal to -209 inches and no Core Spray Subsystem flow can be restored and maintained greater than or equal to 4750 gpm.

   1.3.3 RPV water level cannot be determined and core damage is occurring.

   1.4 Any condition in opinion of Emergency Director that indicates loss of fuel clad barrier. Examples:

   - 15,000 mrem/hr on SJAE monitor.
   - Non-LOCA with DW Rad Monitor reading greater than 115 Rem/hr.
   - Main Steam Line Radiation Monitor readings greater than or equal to Hi-Hi Alarm Setpoint.

ATTACHMENT 6 TRANSIT TIMES AND EFFECTIVE AGES OF NOBLE GASES AT RECEPTOR SITES (INFORMATION USE)

1. Effective Age is defined as time elapsed (hrs) since shutdown. For off-site locations, effective age of isotopic mixture may be obtained through summarizing following components:
   - Effective age at time of release onset.
   - Transit time from release point to receptor site.
2. CALCULATION OF TRANSIT TIME FROM RELEASE POINT TO RECEPTOR LOCATION
   - Estimate downwind distance (miles) to receptor location.
   - Divide distance in miles by 100 meter level (if release from ERP) or 10 meter (if release not from ERP) wind speed (mph) to determine plume transit time.

(1) RECEPTOR SITE DOWNWIND DISTANCE (miles)	(2) 100 METER OR 10 METER LEVEL WIND SPEED (as appropriate) (mph)	(3) PLUME TRANSIT TIME (hrs) (1) ÷ (2)

DETERMINATION OF EFFECTIVE AGES AT RECEPTOR SITES

(1) EFFECTIVE AGE OF MIXTURE AT TIME OF RELEASE ONSET (hrs)	(2) TRANSIT TIME FROM RELEASE POINT TO RECEPTOR LOCATION (hrs)	(3) EFFECTIVE AGE OF ISOTOPIC MIXTURE AT RECEPTOR LOCATION (hrs) (1) + (2)

Name/Time/Date: ______________________/__________/__________

ATTACHMENT 7 PMIS SYSTEM ACCESS AND USE (INFORMATION USE)

1. PLANT MANAGEMENT INFORMATION SYSTEM (PMIS)

   1.1 PMIS System (PMIS) is set of programs and hardware provided by NPPD that make use of VMS functions and additional peripherals (Data Concentrators) which provides access to plant parameters.

2. PMIS COMPUTERS

   2.1 PMIS computers share common set of peripherals (disk drives, terminals, etc.) and software common to that computers security level.

3. VMS OPERATING SYSTEM

   3.1 The VMS Operating System (VMS) is the host operating system for the PMIS computers. It is a set of programs that interface with computer hardware and peripherals, and allows computers to recognize and process commands.

4. PMIS MODES

   4.1 PMIS has three operational modes, Primary, Primary/Backup, and Backup, and will operate on either computer in one of these three modes. Computer with PMIS operating in either Primary or Primary/Backup Mode is referred to as Primary System and one with PMIS operating in Backup Mode is referred to as Backup System.

   4.2 Primary and Primary/Backup Modes provide full PMIS capabilities, consisting (in part) of data acquisition and conversion, data display, data archiving, alarm processing, self-monitoring, and many other functions that perform specialized calculations and displays.

   4.3 Backup Mode monitors Primary System, transfers information necessary to keep Backup System files and tables up-to-date, and automatically changes to Primary Mode when a loss of Primary System is detected (referred to as a FAILOVER). Although many functions are available on Backup System, their use is discouraged because lack of real-time data results in display of inaccurate information (CNS-DOSE is an exception).

5. PMIS ACCESS

   5.1 Access to PMIS is gained through various video display terminals and printers, including color graphic Information Display Terminals (IDTs) dedicated exclusively for PMIS access in Control Room, TSC, and EOF.

5.2 IDTs and printers are selectively connected to either computer through networks controlled by PMIS.

6. SCREEN FORMAT

6.1 When terminal is under control of PMIS (instead of VMS), then screen display will be in standard format consisting of four areas: OCA, GGDA, SSA, and FKA.

6.2 Operator Communication Area (OCA) consists of top two lines on screen. Area is generally used to prompt for and receive user inputs and display advisory and warning messages. In addition, some displays that require only one or two lines of screen use OCA for display. Also (though technically not part of OCA), current date and time (updated once a second) is displayed at right side of screen on lines 1 and 2.

6.3 General and Graphic Display Area (GGDA) consists of lines 4 through 47 and is used for most displays. In addition, some displays (chiefly functions requiring significant editing) also prompt for and receive user inputs in GGDA.

6.4 SPDS Status Area (SSA) consists of lines 45 through 48 and contain four boxes that represent (by color code) status of Safety Parameter Display System (SPDS), which is software system monitoring selected plant parameters and determines overall plant safety status.

6.5 Function Key Area (FKA) consists of bottom two (50 and 51) lines of screen. FKA is used to indicate which of definable function keys are enabled. It indicates which mode PMIS is in, Plant Mode, and whether or not a PMIS "event" has occurred.

7. SCREEN-COPY FUNCTION

7.1 Screen-copy function is activated by **PRESSING** HARD COPY key. It provides full screen reproduction in color on printer located in same general area as terminal.

8. PRINTER

8.1 Printers are connected to a specific computer and are generally accessed when a "...PRINT..." option is selected and a "logical name" is entered.

9. IDE FIELD

9.1 User input to PMIS programs is through an open IDE (Interactive Data Entry) field on terminal. Open IDE field is denoted by yellow box that appears in OCA or GGDA area. Anything typed on keyboard will be echoed in box. Erasing or back-spacing is accomplished with DEL key. All entries into an IDE field must be terminated by **PRESSING** ENTER key unless field is overfilled or a function key is pressed (terminal automatically adds a carriage return character in those cases).

10. TURN-ON-CODE

10.1 Turn-On-Code (TOC) is mechanism by which commands are issued to PMIS. This is one to eight character code interpreted by PMIS and corresponding command is issued.

11. PMIS DATABASE

11.1 All plant parameters (or additional data based on plant or PMIS parameters) processed by PMIS SYSTEM are defined in PMIS DATABASE, which is a file that specifies origin of data, frequency at which it is processed, type of processing to be performed, etc. Each parameter is referred to as a "point" and is identified by a one to eight character name or POINT-ID (PID).

12. PMIS DATA PROCESSING

12.1 Some PMIS points are processed by scanning plant sensors (through Data Concentrator) while others are calculated based on values of previously processed points or PMIS parameters. All point values are then assigned a quality code stored in Current Value Table (CVT).

12.2 Data in CVT is considered to be "real-time" and representative of current plant and system conditions.

12.3 At regular intervals (and other special circumstances), point values are also stored in an Archive File, which provides ~ 24 hours of on-line historical information.

13. PMIS DATA ACCESS

13.1 All point values in CVT and Archive File are accessed by the POINT-ID.

14. QUALITY CODES

14.1 Quality Code, assigned when point values are assigned, represents general status and "health" of point, and determines how it is used by PMIS programs. Following is list of PMIS quality codes and related information.

CODE	DESCRIPTION	COLOR	HEALTH
UNK	Value unknown - not yet processed	White	Bad
DEL	Processing has been disabled	Magenta	Bad
NCAL	Value cannot be calculated	Magenta	Bad
INVL	Data concentrator error	Magenta	Bad
RDER	Data concentrator error	Magenta	Bad
OIC	Data concentrator error	Magenta	Bad
BAD	Outside instrument range	Magenta	Bad
STAG	Point failed stagnation check	Magenta	Bad
UDEF	Undefined (spare)	Magenta	Bad
REDU	Fails redundant point check	Magenta	Bad
HALM	Above high alarm limit	Red	Good
LALM	Below low alarm limit	Red	Good
HWRN	Above high warning limit	Yellow	Good
LWRN	Below low warning limit	Yellow	Good
ALM	State/Change-of-State alarm	Red	Good
SUB	Value has been substituted	Blue	Good
DALM	Alarm checking has been disabled	Green	Good
INHB	Alarm inhibited by cut-out point	Green	Good
GOOD	Passes all other checks	Green	Good

14.2 Not listed above is quality code OSUB (Operator Substituted), which is treated same as SUB, and indicates value was substituted within program. OSUB is not used in CVT.

15. PMIS LOGIN

15.1 IF windows desktop displayed,
THEN **START** IDT software.

15.2 IF Connection and Data Server Information dialog box displayed,
THEN **PRESS** Attempt Connection button.

15.3 IF current date and time displayed in OCA and is being updated about once a second,
THEN **PERFORM** following:

NOTE – Password information will not be displayed for CDA IDT as it is entered automatically.

15.3.1 IF "ENTER PASSWORD..." displayed on line 2 for CDA IDT,
THEN IDT is to be re-booted.

15.3.2 IF IDT is business computer,
THEN **ENTER** designated password and **PRESS** ENTER key.

15.3.3 IF "SELECT FUNC. KEY OR TURN ON CODE..." and open IDE field displayed on line 2,
THEN IDT is logged into PMIS. No further action is necessary.

15.3.4 IF display is operating (such as DOSE program),
THEN **PRESS** CANC key.

15.3.5 IF terminal does not respond or does not meet any of above criteria,
THEN **PRESS** reset key once and **LOG** on again.

15.3.6 IF neither of above criteria met or specified sequence of events does not occur,
THEN **CONTACT** Information Technology (IT) Department for assistance.

16. ACTIVATING A TURN-ON-CODE

16.1 IF display currently operating in area of screen desired, TOC requires,
THEN **PRESS** CANC key.

16.2 WHEN "SELECT FUNC. KEY OR TURN ON CODE..." displayed followed by open IDE field,
THEN **ENTER** one of following:

16.2.1 TOC (i.e., "GROUP" -- activates Group Display Program; program will then prompt user to select menu option).

16.2.2 TOC followed by space and optional text (i.e., "PLOT ARM1" -- activates Real-Time Plot Program and plots group "ARM1" without further user input; note optional text is recognized by only selected TOCs).

16.2.3 **PRESS** one of programmable function keys on right hand key pad or top row of function keys (i.e., blue "GROUP DISP" key -- functions same as first example).

16.3 **REFER** to FKA for function keys enabled and their descriptions. **USE** other options as provided by each program.

16.4 To **EXIT** a program, **USE** specified exit option (if provided) or **PRESS** CANC function key.

17. DETERMINING TO WHICH SYSTEM A TERMINAL IS CONNECTED

17.1 PMIS System to which terminal is connected is indicated by "CONSOLE =..." on bottom line of FKA as follows:

- CONSOLE = PRIMARY - Connected to Primary System operating in Primary Mode.
- CONSOLE = PRIM/BAC - Connected to Primary System operating in Primary/Backup Mode.
- CONSOLE = BACKUP - Connected to Backup System.
- CONSOLE = UNKNOWN - PMIS is in transition or unknown state.

ATTACHMENT 8 INFORMATION SHEET (INFORMATION USE)

1. PURPOSE℗[1]©[6]

 1.1 Perform dose assessment using CNS-DOSE computer program during emergency event resulting in off-site radiological release.

2. PRECAUTIONS AND LIMITATIONS

 2.1 When adjusting CNS-DOSE with field sample results, CNS-DOSE shall not generate dose assessment projections based on field sample values that are lower than projected dose values of CNS-DOSE.

 2.2 Control Room is not required to perform Attachment 2 for correlating field sample results to CNS-DOSE projection results as specified in NUREG 0654, Revision 1; Pages 2 through 9.

3. DISCUSSION

 3.1 CNS-DOSE is programmed to perform dose projections from Multiple Source Releases. CNS-DOSE calculates projected off-site dose and dose rates based upon aggregate releases from both ground level and elevated release sources.©[8]

 3.2 CNS-DOSE Computer program is software application operated on PMIS computers and is Primary method of dose projection.

 3.3 Following equipment and materials are needed to perform CNS-DOSE and are located in Dose Assessment Area of EOF and TSC:

 - Computer terminals.
 - Computer printers.
 - PMIS data healthy.

 3.4 CNS-DOSE makes use of current meteorological and radiological data from PMIS and manually entered data to perform dose projection for area surrounding CNS.

 3.5 CNS-DOSE program will abbreviate words such as release (rel), wind direction (DIR) in degrees (deg), 100M tower (meter), Standby Gas Treatment (SGT), etc.

3.6 When performing CNS-DOSE, data can be entered to perform "what if" or "look ahead" projections. Data used for decision making purposes must be accurate for release point being evaluated (i.e., how does release go from fuel to release point).

3.7 When CNS-DOSE program is started or NEw Sample option is selected, then new data will be loaded into program, updating previously loaded information and resetting manually entered information.

3.8 Where possible, data used is from same source as used by computer programs.

3.9 Initial CNS-DOSE projections are based upon assumed radionuclide concentrations until actual concentrations have been measured. Off-site sample results from Field Monitoring Teams are used to determine dose correction factor which may be applied to adjust CNS-DOSE program.

3.9.1 Attachment 2 provides EOF Dose Assessment personnel with means of correlating Field Monitoring Team iodine concentration data with CNS DOSE projected iodine concentration.

3.9.2 Radiological Assessment Supervisor (RAS) compares Field Monitoring Team sample results to CNS-DOSE reports that closely correspond with time of release from plant considering transit time from release point to field sample location.

3.9.3 Correlation methodology as described in Attachments 2 and 3 provide EOF Dose Assessment personnel with means of correlating Field Monitoring Team iodine concentration data with CNS-DOSE projected iodine concentration. Such a correlation is necessary to determine if initial Protective Action Recommendations (PARs) were adequate to protect health and safety of public.

3.9.4 For adjustment purposes, Field Monitoring Team air sample results are assumed to be plume centerline results. When adjusting for field team results, CNS-DOSE program data needs to reflect conditions in existence at time plume segment being sampled was being released. New data will need to be manually adjusted and entered into CNS-DOSE to reflect correct historical data to yield accurate field team-corrected dose assessment results.

4. RECORDS

4.1 Attachment 3 from actual events shall be forwarded to EP Manager within 5 working days of their completion (quality record upon completion).

5. REFERENCES

5.1 TECHNICAL SPECIFICATIONS

5.1.1 Technical Specification Bases B.3.3.6.1 (2.d), Main Steam Line Radiation - High.

5.2 CODES AND STANDARDS

5.2.1 EPA 400-R-92-001, May 1992, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents.

5.2.2 Health Physics Journal, November 1981, Noble Gas Dose Rate Conversion Factors.

5.2.3 ICRP 59, Working Breathing Rate.

5.2.4 NRC Regulatory Guide 1.23, Revision 1, March 2007, Meteorological Programs for Nuclear Power Plants.

5.2.5 NRC Regulatory Guide 1.109, Revision 1, October 1977, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for Purpose of Evaluating Compliance with 10CFR50, Appendix I, Iodine Inhalation Dose Factors.

5.2.6 NRC Regulatory Guide 1.111, July 1977, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors.

5.2.7 NRC Regulatory Guide 1.145, August 1979, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.

5.3 DRAWINGS (MAPS)

5.3.1 Cooper Nuclear Station 50 Mile Emergency Planning Zone, 50 Mile Radius.

5.3.2 NPPD Drawing CNS-MI-03, Preselected Radiological Sampling and Monitoring Points in the Vicinity of Cooper Nuclear Station, 10 Mile Radius.

5.3.3 NPPD Drawing CNS-MI-102, Atmospheric Dispersion Model (EPM2) Special Receptor Points, 10 Mile Radius.

5.3.4 NPPD Drawing 2.2 (P3-A-45), Cooper Nuclear Station Site and Property Boundary, 1 Mile Radius.

5.4 PROCEDURES

5.4.1 Computer System Operating Procedure 2.6.3PMIS, PMIS Computer System Operation and Outage Recovery.

5.4.2 Emergency Plan Implementing Procedure 5.7.1, Emergency Classification.

5.4.3 Emergency Plan Implementing Procedure 5.7.16, Release Rate Determination.

5.4.4 Emergency Plan Implementing Procedure 5.7.17.1, Dose Assessment (Manual).

5.4.5 Emergency Plan Implementing Procedure 5.7.20, Protective Action Recommendations.

5.5 MISCELLANEOUS

5.5.1 Engineering Evaluation EE 02-056, Elimination of Meteorological Instrumentation System Strip Chart Recorder References.

5.5.2 NEDC 99-034, Control Room, EAB, and LPZ Doses Following a CRDA.

5.5.3 NEDC 02-004, Estimation of Steam Jet Air Ejector Radiation Monitor, RMP-RM-150A (B), Readings Following a 1% Fuel Clad Release (Degraded Core) in Reactor Coolant System.

5.5.4 NEDC 02-009, Estimation of Primary Containment High Range Monitor, RMA-RM-40A (B), Readings Following 1% Clad Failure in RCS Under Non-LOCA Conditions.

5.5.5 NEDC 02-20, Estimation of Reactor Coolant System Dose Equivalent I-131 Concentration Following a 1% Fuel Clad Failure (Degraded Core) Under Non-LOCA Conditions.

5.5.6 NEDO-31400, Safety Evaluation for Eliminating BWR MSIV Closure Function and Scram Function for MSL Rad Monitors.

5.5.7 NEI 99-01, Revision 5, SER TAC Number ME0840.

5.5.8 SAIC Document Number 502-85500107-72, PMIS Operator's Manual.

5.5.9 Software Alteration 5000710, Modify CNS-DOSE For Multi-Point Release, December 2014.©[8]

5.5.10 ℗[1] TIP Action Plan 5.2.2.1, Revision 1, Action 1. Major revision 6/28/02 to clarify Control Room tasks. Affects entire procedure (Attachment 8, Step 1 flagged).

5.6 NRC COMMITMENTS

5.6.1 ©[2] NRC Commitment NLS91000613-05. NPPD Response to IR 91-12, Assure the correct input for the Standby Gas Treatment System is selected. Commitment affects Attachment 1, NOTE prior to Step 3.7 and Step 3.7.1.

5.6.2 ©[3] NRC Commitment NLS91000613-06. NPPD Response to IR 91-12, Facilitate correlation of field team data with dose projections and facilitate verification of protective action recommendations. Commitment affects Attachments 2, 3, and 4.

5.6.3 ©[4] NRC Commitment NSD921204-07, IR 92-14 Corrective Actions. Cladding Damage Criterion was clarified. Commitment affects Attachment 1, Step 3.9.1, and Attachment 5.

5.6.4 ©[5] NRC Commitment NLS8900460-03 and 04 NPPD Response to IR 89-35. Commit 3 was to stress importance of selecting correct setting for "core degraded" input to dose assessment computer models and Commit 4 was to require coordination between Engineering, Operations, and Health Physics personnel in determining "core degraded" status. Commitment affects Attachment 1, Step 3.9.2, and Attachment 5, NOTE before Step 1.

5.6.5 ©[6] NRC Commitment 811217-01-07. NPPD Response to IR 81-13, Develop a Functional Procedure for Dose Assessment. Commitment affects entire procedure (Attachment 8, Step 1 flagged).

5.6.6 ©[8] NRC Commitment NLS2013056-01, Revise offsite Dose Assessment computer methodology to include capability to automatically sum selected assessment results using a revised computer code. Commitment affects Attachment 1 and Attachment 8, Steps 3.1 and 5.5.9.

5.6.7 ©[9] NRC Commitment NLS2014035-02, to maintain the capability for classifying fuel damage events at the Alert level threshold for CNS at radioactivity levels of 300 µCi/cc dose equivalent iodine. Commitment affects Attachment 5, Step 1.1.

Enclosure 6

Emergency Plan Implementing Procedure 5.7.20, Revision 27

COOPER NUCLEAR STATION

Operations Manual

Emergency Preparedness

EMERGENCY PLAN IMPLEMENTING PROCEDURE

5.7.20

PROTECTIVE ACTION RECOMMENDATIONS

Level of Use:	**MULTIPLE**
Quality:	**QAPD RELATED**
Effective Date:	**3/28/16**
Approval Authority:	**ITR-RDM**
Procedure Owner:	**EMER PREP DRILL COORD**

TABLE OF CONTENTS

1. ENTRY CONDITIONS (INFORMATION USE)

   1.1 Projected Total Effective Dose Equivalent (TEDE) or Committed Dose Equivalent (CDE) dose per EPIP 5.7.17 or 5.7.17.1 exceed 1 rem TEDE or 5 rem CDE at or beyond boundary of Owner Controlled Area (OCA).

   1.2 GENERAL EMERGENCY (GE) declared based on plant conditions.

2. INSTRUCTIONS (REFERENCE USE)

   2.1 **USE** Table 1 to select appropriate guidance:

Table 1	
CONDITION	GO TO
IF EOF not available and Shift Manager is Emergency Director (ED)	Attachment 1
IF EOF activated with on-call ED, RAS, and RCM	Attachment 2

ATTACHMENT 1 PAR DETERMINATION FROM CONTROL ROOM (INFORMATION USE)

1. Dose Assessor **USE** Table 1 to select appropriate section based on dose assessment method:

Table 1	
DOSE ASSESSMENT METHOD	GO TO
EPIP 5.7.17, CNS-DOSE	Section 2
EPIP 5.7.17.1, Hand Calculated Dose Assessment	Section 3

2. Dose Assessor **OBTAIN** CNS-DOSE Assessment Report and **REVIEW** PARs.

2.1 **USE** Attachments 3 and 4 to check CNS-DOSE PAR determination.

2.1.1 IF projected dose at 2 miles exceeds 1 rem TEDE or 5 rem CDE, THEN **ADD** downwind sectors out to 5 miles to PAR.

2.1.2 IF projected dose at 5 miles exceeds 1 rem TEDE or 5 rem CDE, THEN **ADD** downwind sectors out to 10 miles to PAR.

2.1.3 IF projected dose at 10 miles exceeds 1 rem TEDE or 5 rem CDE, THEN **PERFORM** following:

2.1.3.1 **RECOMMEND** evacuation out to 15 miles in affected sectors.

2.1.3.2 **INCLUDE** in Remarks Section of CNS Notification Report Form that "Projected Dose at or beyond 10 miles exceeds evacuation criteria, **RECOMMEND** evacuation out to 15 miles in affected sectors".

2.2 **USE** Attachment 5 to check appropriate downwind sectors selected for PARs based on wind direction.

2.3 **PROVIDE** CNS-DOSE Assessment Report and PARs to Shift Manager-Emergency Director (ED) for review.

2.3.1 IF Shift Manager-ED approval of dose assessment and PARs is separate from Notification Report, THEN **RECORD** approval of PARs listed on CNS-DOSE Assessment Report in Operations Log.

2.4 **PROVIDE** CNS-DOSE Assessment Report to Shift Communicator to complete CNS Notification Form per EPIP 5.7.6.

2.5 **PROCEED** to Step 4.

3. Shift Communicator **PERFORM** manual CNS Notification Report per EPIP 5.7.6.

3.1 Dose Assessor **COMPLETE** hand calculated dose assessment per EPIP 5.7.17.1.

3.2 Shift Communicator and Dose Assessor **PERFORM** Attachments 3 and 4 to determine PARs:

3.2.1 IF projected dose at 2 miles exceeds 1 rem TEDE or 5 rem CDE, THEN **ADD** downwind sectors out to 5 miles to PAR.

3.2.2 IF projected dose at 5 miles exceeds 1 rem TEDE or 5 rem CDE, THEN **ADD** downwind sectors out to 10 miles to PAR.

3.2.3 IF projected dose at 10 miles exceeds 1 rem TEDE or 5 rem CDE, THEN **PERFORM** following:

3.2.3.1 **RECOMMEND** evacuation out to 15 miles in affected sectors.

3.2.3.2 **INCLUDE** in Remarks Section of CNS Notification Report Form that "Projected Dose at or beyond 10 miles exceeds evacuation criteria, **RECOMMEND** evacuation out to 15 miles in affected sectors".

3.2.4 **USE** Attachment 5 to determine appropriate downwind sectors selected for PARs based on wind direction.

3.3 **PROVIDE** hand calculated dose assessment and PARs to Shift Manager-Emergency Director (ED) for review.

3.3.1 IF Shift Manager-ED approval of dose assessment and PARs is separate from Notification Report, THEN **RECORD** approval of PARs listed on CNS-DOSE Assessment Report in Operations Log.

3.4 Shift Communicator **COMPLETE** CNS Notification Form per EPIP 5.7.6 with hand calculated dose assessment results and PARs.

CAUTION – Emergency Director (ED) cannot delegate responsibility for approving PARs.

4. Shift Manager-ED **APPROVE** completed CNS Notification Report per EPIP 5.7.6 including dose assessment and PARs.

NOTE – Initial Notification to State/Local Agencies shall be completed within 15 minutes after determination of PARs or change in PARs.

5. Shift Communicator **PERFORM** notification including PARs to State and Local Agencies per EPIP 5.7.6.

6. Dose Assessor **USE** Attachment 5 during wind shift to forecast affected downwind sectors.

7. Dose Assessor **CONTINUE** to assess critical parameters affecting PARs.

   7.1 IF any one of following critical parameters change:
   - Combination of ERP or BLDG effluent release sources.
   - Degraded core criteria.
   - MET data; wind shift (in degrees), wind speed and precipitation, when change can add sector(s) to PARs.

   THEN **EVALUATE** PARs for possible upgrade.

8. Dose Assessor **COMPARE** new PARs to PARs previously transmitted to off-site authorities.

9. WHEN GE terminated or off-site dose below 1 rem TEDE and 5 rem CDE at or beyond boundary of Owner Controlled Area (OCA),
   THEN **EXIT** procedure.

ATTACHMENT 2 PAR DETERMINATION FROM EOF (INFORMATION USE)

1. Radiological Assessment Supervisor (RAS) **ASSIGN** critical parameters for dose assessment personnel in EOF to monitor continuously during release.

2. Radiological Control Manager (RCM) **ENSURE** following:
   - Critical parameters are assigned and monitored continuously during release.
   - Follow-up dose assessments and PAR determinations are performed when critical parameters are met or exceeded.

3. **USE** Table 1 and select appropriate section based on dose assessment method.

Table 1	
DOSE ASSESSMENT METHOD	GO TO
EPIP 5.7.17, CNS-DOSE	Section 4
EPIP 5.7.17.1, Hand Calculated Dose Assessment	Section 5

4. RCM **PERFORM** following:

   4.1 **REVIEW** CNS-DOSE Assessment Report.

   4.2 **VERIFY** CNS-DOSE PARs listed on report are correct for plant and radiological conditions.

   4.2.1 **VERIFY** PAR selection per Attachments 3 and 4.

   4.2.2 **VERIFY** appropriate downwind sectors selected for PARs based on wind direction per Attachment 5.

   4.3 IF projected dose at 2 miles exceeds 1 rem TEDE or 5 rem CDE, THEN **ADD** downwind sectors out to 5 miles to PAR.

   4.4 IF projected dose at 5 miles exceeds 1 rem TEDE or 5 rem CDE, THEN **ADD** downwind sectors out to 10 miles to PAR.

   4.5 IF projected dose at 10 miles exceeds 1 rem TEDE or 5 rem CDE, THEN concurrently **PERFORM** Section 6.

4.6 **PROVIDE** CNS-DOSE Assessment Report and PARs to ED for review.

4.6.1 IF ED approval of dose assessment and PARs is separate from Notification Report,
THEN **RECORD** approval of PARs listed on CNS-DOSE Assessment Report in facility log.

4.7 **PROVIDE** EOF Off-Site Communicator with CNS-DOSE Assessment Report and PARs for completion of CNS Notification Report per EPIP 5.7.6.

4.8 RCM and EOF Off-Site Communicator **PROVIDE** ED with completed CNS-DOSE Assessment Report and PARs for review.

4.9 **PROCEED** to Step 7.

5. IF dose assessment generated by hand calculations per EPIP 5.7.17.1,
THEN RCM **PERFORM** following:

5.1 **REVIEW** hand calculated dose assessment per EPIP 5.7.17.1 from Dose Assessment personnel.

5.2 Concurrently **PERFORM** Attachments 3 and 4 to determine PARs.

5.3 IF projected dose at 2 miles exceeds 1 rem TEDE or 5 rem CDE,
THEN **ADD** downwind sectors out to 5 miles to PAR.

5.4 IF projected dose at 5 miles exceeds 1 rem TEDE or 5 rem CDE,
THEN **ADD** downwind sectors out to 10 miles to PAR.

5.5 IF projected dose at 10 miles exceeds 1 rem TEDE or 5 rem CDE,
THEN concurrently **PERFORM** Section 6.

5.6 **USE** Attachment 5 to determine appropriate downwind sectors selected for PARs based on wind direction.

5.7 **PROVIDE** hand calculated dose assessment results and PARs to ED for review.

5.7.1 IF ED approval of dose assessment and PARs is separate from Notification Report,
THEN **RECORD** approval of PARs listed on CNS-DOSE Assessment Report in facility log.

5.8 **PROVIDE** Off-Site Communicator with hand calculated dose assessment results and PARs for completion of CNS Notification Report per EPIP 5.7.6.

5.9 RCM and EOF Off-Site Communicator **PROVIDE** ED with completed manual CNS Notification Report for review.

5.10 **PROCEED** to Step 7.

6. **DIRECT** calculation of off-site dose estimate to determine distance beyond 10 mile EPZ at which projected or actual dose is less than or equal to 1 rem TEDE and less than or equal to 5 rem CDE.

6.1 Dose Assessment Staff **OBTAIN** estimated dose at 10 mile point from Dose Assessment Report or manual calculation form per EPIP 5.7.17.1.

6.2 **USE** dose value at 10 miles to determine reduction factor required to equate to dose projection of 1 rem TEDE or 5 rem CDE.

6.2.1 For example, IF dose at 10 miles is projected to be 2 rem TEDE, THEN reduction of one half is required.

6.3 **SELECT** isopleth based on release source and atmospheric stability class.

6.4 **USE** correct isopleth to determine χ/Q value from centerline column at 10 miles.

6.5 **DETERMINE** required χ/Q based on reduction factor and find distance related to that value.

6.5.1 Continuing example from Step 6.2.1:

6.5.1.1 IF χ/Q centerline value at 10 miles is 4.51E-8 and reduction factor of one half is required,
THEN at what distance is χ/Q value less than or equal to 2.255E-8?

Answer: IF χ/Q value at 15 miles was 2.29E-8 and at 16 miles is 2.06E-8,
THEN PAR should extend to 16 miles to ensure PARs issued for areas expected to receive TEDE dose of 1 rem or more.

6.6 RCM **RECOMMEND** PARs to distance determined by Dose Assessment Staff in affected sectors.

6.7 RCM **INCLUDE** in Remarks Section of CNS Notification Report Form that "Projected Dose at or beyond 10 miles **EXCEEDS** evacuation/shelter criteria. **RECOMMEND** evacuation/shelter out to _______ miles in Sectors ___________".

CAUTION – Emergency Director (ED) cannot delegate responsibility for approving PARs.

7. ED **APPROVE** completed CNS Notification Report Form per EPIP 5.7.6 including dose assessment results and PARs.

NOTE – State, Federal, or Local Dose Assessment personnel may be present in EOF and conducting independent dose assessments. These personnel should be consulted or results reviewed in process of making a PAR.

8. IF Nebraska and/or Missouri State Representatives are present in EOF, THEN ED will **MAKE** official communication of PAR to authorities verbally and should **PROVIDE** written information concurrently via Cooper Nuclear Station Notification Report per EPIP 5.7.6.

9. IF Nebraska and/or Missouri State Representatives are not present in EOF, THEN official **COMMUNICATION** of PAR is via CNS Notification process performed per EPIP 5.7.6.

10. Off-Site Communicator **PERFORM** notifications including dose assessment and PARs to State and Local Agencies per EPIP 5.7.6.

11. RCM and RAS **USE** Attachment 5 during wind shift to forecast affected downwind sectors.

12. RAS and Dose Assessment Team **CONTINUE** to assess critical parameters affecting PARs.

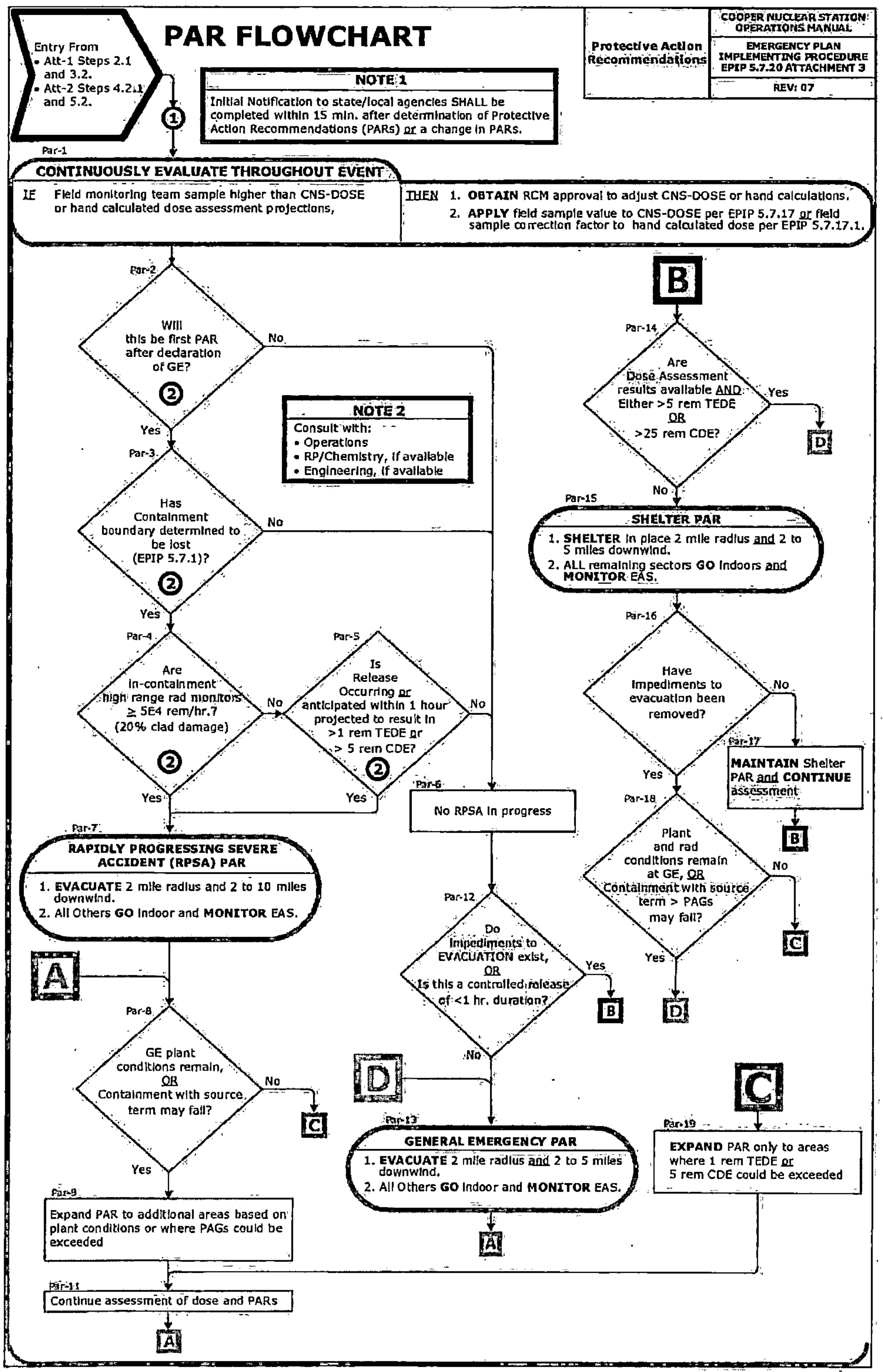
    12.1 IF any one or more of following critical parameters change:
    - Combination of ERP or BLDG effluent release sources.
    - Degraded core criteria.
    - MET data; wind shift (in degrees), wind speed and precipitation, when change can add sector(s) to PARs.

    THEN **EVALUATE** PARs for possible upgrade.

13. RCM **COMPARE** new PARs to PARs previously transmitted to off-site authorities.

14. WHEN GE terminated or off-site dose below 1 rem TEDE and 5 rem CDE at or beyond boundary of Owner Controlled Area (OCA),
THEN **EXIT** procedure.

ATTACHMENT 3 PROTECTIVE ACTION RECOMMENDATION FLOWCHART (REFERENCE USE)



ATTACHMENT 4 FLOWCHART DECISION BASIS (INFORMATION USE)

1. Entry Arrow: Flowchart assumes entry to procedure is met which is either of following:

    1.1 Projected Total Effective Dose Equivalent (TEDE) exceeds 1 rem or Committed Dose Equivalent (CDE) exceeds 5 rem CDE at or beyond boundary of Owner Controlled Area (OCA).

    1.2 GENERAL EMERGENCY (GE) based on plant conditions.

    1.3 In addition entry to Attachment 3, Protective Action Recommendation (PAR) flowchart is from either Attachment 1 for PAR determinations from Control Room, or Attachment 2 during PAR determination from EOF.

2. NOTE 1: Note to emphasize that once PAR is determined and approved by Emergency Director (ED), whether initial PAR or an upgraded PAR, there is a requirement that State and Local Agencies are to be notified within 15 minutes. This same requirement is also provided in EPIP 5.7.6 which governs notification but is reinforced here.

    2.1 Note is provided early in flowchart to give sense of urgency to expedite PAR confirmation or determination especially when CNS-DOSE is not available.

3. Step Par-1 Override: *Continuously Evaluate Throughout Event.* Override step contains one conditional based action. Override action is applicable during performance of entire flowchart and performed if conditional statement portion is met.

    3.1 Conditional statement column describes condition when Field Monitoring Team field sample readings are higher than dose assessment projection results. This is non-conservative and requires action to raise dose assessment projection, either through CNS-DOSE or hand calculation methods, to make conservative or higher than Field Team samples.

    3.2 Action column requires RCM prior approval and refers to EPIP 5.7.17 or 5.7.17.1 to perform correction to dose assessment.

4. **DETERMINE** if Rapidly Progressing Severe Accident (RPSA) is occurring.

   4.1 IF these criteria cannot be immediately confirmed,
   THEN **ASSUME** RPSA is not taking place.

   4.2 Step Par-2: *Will this be first PAR made after declaration of GE?* This question is establishing whether there is an existing PAR in effect and being executed as a result of either plant conditions at GE or radiological release driving off-site dose. The basis for applying RPSA extended PARs assumes no existing PAR in effect with rapidly degrading plant conditions that warrant RPSA extended PARs.

      4.2.1 IF PARs already in effect,
      THEN RPSA extended PARs are not warranted.

      4.2.2 Note 2: This note is to emphasize that responding to RPSA decision step may require consultation between other members of ERO including Operations, Radiological Protection or Chemistry personnel, if available; and Engineering personnel, if available.

   4.3 Step Par-3: *Has Containment Boundary been determined to be lost per EPIP 5.7.1, Attachment 3, Table F-1?* Potential losses do not meet this criterion. The intention is to anticipate significant radiological release and off-site dose due to loss of containment meeting assumptions that warrant extended RPSA default PARs.

      4.3.1 IF containment is intact,
      THEN applying RPSA extended PARs are not warranted.

      4.3.2 Note 2: This note is to emphasize that responding to RPSA decision step may require consultation between other members of ERO including Operations, Radiological Protection or Chemistry personnel, if available; and Engineering personnel, if available.

   4.4 Step Par-4: *Are in-containment high range radiation monitors indicating greater than or equal to 5E4 rem/hr?* This corresponds to equivalent of about 20% clad damage. This step provides a quick assessment as to state of reactor core and fuel integrity.

      4.4.1 IF in-containment monitors indicate this level of fuel damage,
      THEN applying RPSA extended PARs is warranted in anticipation of eventual extremely high radiological release rates and resulting off-site dose.

4.4.2 IF in-containment monitors do not indicate this level of fuel damage,
THEN applying RPSA extended PARs is not warranted unless there is an impending radiological release that will exceed EPA limits anticipated within one hour as reflected in Step Par-5.

4.4.3 Note 2: This note is to emphasize that responding to RPSA decision step may require consultation between other members of ERO including Operations, Radiological Protection or Chemistry personnel, if available; and Engineering personnel, if available.

4.5 Step Par-5: *Is release occurring or anticipated within 1 hour projected to result > 1 rem TEDE or > 5 rem CDE?* The basis of applying RPSA extended PARs is justified if a radiological release is occurring or will occur within an hour that will exceed EPA limits of 1 rem TEDE or 5 rem CDE. This is appropriate regardless of level of fuel damage as indicated by in-containment monitors.

4.5.1 IF there is no radiological release occurring or anticipated within one hour expected to exceed EPA limits of 1 rem TEDE or 5 rem CDE,
THEN RPSA extended PARs are not warranted.

4.5.2 Note 2: This note is to emphasize that responding to RPSA decision step may require consultation between other members of ERO including Operations, Radiological Protection or Chemistry personnel, if available; and Engineering personnel, if available.

4.6 Step Par-6: *No RPSA in progress.*

4.6.1 IF decisions of Step Par-2, Par-3, Par-4, and Par-5 are NO,
THEN RPSA is not occurring and extended RPSA PARs are not warranted. However, depending on answers to decisions in subsequent Step Par-12, GE default PARs may be warranted.

5. Step Par-7: *Rapidly Progressing Severe Accident (RPSA) PAR.*

   5.1 IF decisions of Step Par-2, Par-3, and either Step Par-4 or Par-5, are YES,
   THEN RPSA is occurring and extended RPSA PAR is to be issued. RPSA PAR requires everyone within two mile radius of plant and everyone in downwind plume pathway affected sectors up to ten miles distance from plant to evacuate. Decision does not consider impediments to evacuation or controlled release considerations. This is an additional five miles beyond default PAR for General Emergencies due to severe accident conditions of loss of containment with significant fuel damage or extreme radiological releases. Remaining sectors of ten mile EPZ are directed to go indoors and monitor EAS for further instructions.

6. Step Par-8: Step begins flowchart loop of continual re-assessment of off-site dose and PARs following issuance of extended RPSA PAR. Step is entered from Step Par-11 whose action is to continue to assess dose and PARs for potentially expanding them to address changes in MET and rad conditions or changes in plant barriers.

   6.1 *GE conditions remain* is asking if plant conditions have changed and if conditions for GE are still met.

      6.1.1 IF GE plant conditions continue to be in effect,
      THEN expansion to additional areas or sectors where PAGs may be exceeded is warranted.

   6.2 *Containment with source term may fail* is asking:

      6.2.1 IF containment is threatened such that it may fail due to continuing degraded or adverse plant conditions (greater than 56 psig or greater than deflagration concentrations) with a release of a source term that would result in exceeding EPA limits of 1 rem TEDE or 5 rem CDE,
      THEN expansion of existing PAR to additional areas or sectors where PAGs may be exceeded is warranted.

   6.3 IF answer is YES,
   THEN flow path directs to Step Par-9 to expand PAR affected areas.

   6.4 IF answer is NO,
   THEN flow path directs to Step Par-19 action to expand current PAR only to areas where EPA limits of 1 rem TEDE or 5 rem CDE are exceeded.

7. Step Par-9: *Expand PAR to additional areas based on plant conditions or where PAGs could be exceeded.* This step directs PARs be expanded to new areas only if EPA limits of 1 rem TEDE or 5 rem CDE are projected to be exceeded in those areas. This expansion could be warranted due to degraded plant conditions, either containment integrity or amount of fuel damage, or from changes in rad or MET conditions affecting off-site dose.

8. Step Par-10: N/A

9. Step Par-11: *Continue assessment of dose and PARs.* This step directs assessment of dose projections as well as PARs to ensure health and safety of public is maintain during changing conditions. This step is entered from two flow paths both of which have expanded existing PARs based on changes to plant conditions or off-site dose to ensure PARs are effectively issued. The step directs performer to re-enter flow path at Step Par-8 to re-address decision regarding plant conditions and containment with a source term that could cause off-site dose to exceed EPA limits of 1 rem TEDE or 5 rem CDE. This flow path loop of reassessing for changes and possible expansion of existing PARs continues until answer to Step Par-8 is NO or entry to PAR flowchart is no longer met.

10. Step Par-12: This decision step is effectively assessing whether evacuation is safest or practical strategy for protecting health and safety of public from radiological exposure verses sheltering in-doors.

    10.1 *Do impediments to EVACUATION exist?* Impediments are those conditions that would put evacuees in greater danger because of evacuation process than that posed by projected plume exposure.

    10.1.1 These would typically be based on road closures determined by State Highway Patrol or other Local or State Agency, or due to natural phenomena such as a blizzard. CNS is informed of these impediments in advance by off-site authorities.

    10.1.2 Impediments due to road conditions need only be considered for population zones that have not already been evacuated.

    10.1.3 IF affected areas being considered for PAR have already been evacuated due to condition causing impediment,
    THEN answer to question should be NO.

10.1.4 In event of limited infrastructure impact due to an event, such as closing an individual evacuation route with other safe routes available, evacuation should not be considered impeded if area would be able to be evacuated via alternate routes within 2 hour period of time.

10.1.5 General Emergency condition based on Hostile Action is considered an impediment to evacuation. Impediment is effective until Off-Site Incident Commander (typically Nemaha County Sheriff) has determined conditions are safe for evacuations to proceed and communicated to CNS representatives either at Incident Command Post (ICP) or ED in EOF.

10.2 *Is this a controlled release of < 1 hr. duration?* This question is asking if radiological release is controlled and of relatively short duration (puff release), then in-door sheltering may be effective and prudent to protect public safety verses inherent risks associated with evacuation.

10.2.1 Controlled means that release was initiated by Operator Action and can be terminated by Operator Action. Loss of ability to control release warrants reassessment of condition and re-evaluation of PAR.

10.2.2 One hour is selected as maximum allowed release duration based on expected evacuation time under adverse conditions.

10.3 IF impediments to evacuation exist or if rad release is controlled and short duration,
THEN flow path directs that sheltering be performed until impediments to evacuation no longer exist or dose assessment exceeds five times EPA limits of 1 rem TEDE or 5 rem CDE.

11. Step Par-13: General Emergency (GE) PAR. Based on no RPSA occurring and Par-12 being answered NO, GE default PAR is made to evacuate 2 mile radius and 2 to 10 miles downwind, and all others go indoors and monitor EAS.

11.1 Following issuance of GE PAR, flow path directs entry to Step Par-8 to begin loop of continual re-assessment of off-site dose and plant conditions including containment and source term that could be released due to fuel damage that may warrant expansion of PAR to additional areas.

12. Step Par-14: *Are Dose Assessment results available AND Either > 5 rem TEDE OR > 25 rem CDE?* Step specifies maximum dose up to which shelter will be recommended instead of evacuation if dose assessment results are available and indicate less than five times EPA limits of 1 rem TEDE or 5 rem CDE.

    12.1 Values are based on maximum dose specified from EPA 400 information on Attachment 6 for an Evacuation PAR.

    12.2 ED should not wait on dose assessment results if they are not readily available.

    12.3 <u>IF</u> projected dose exceeds five times EPA limits at site boundary, <u>THEN</u> flow path directs to Step Par-13 to issue GE PAR which requires evacuation recommendation regardless of evacuation impediments <u>or</u> controlled releases due to anticipated dose.

13. Step Par-15: *Shelter PAR.* With dose assessment less than five times EPA limits, this alarm action step specifies 2 mile radius and 5 miles downwind shelter in place instead of evacuation, and all others go indoors and monitor EAS. Current projected dose and impediments for evacuation are such that sheltering is preferred over evacuation until impediments no longer exist, dose rates change, or state and local Authorities can implement other protective actions for affected populations.

14. Step Par-16: *Have evacuation impediments been removed?* This step reassesses impediments to evacuation discussed in Step Par-12 to determine appropriate decision logic to implement under changing plant conditions.

    14.1 Answering NO means that impediments to evacuation are still present flow path directs to Step Par-17 to maintain current shelter PAR and continue to reassess dose assessment in Step Par-14.

    14.2 Answering YES means impediments to evacuation are removed directing assessment of plant and radiological conditions in Step Par-18 to require change in PAR.

15. Step Par-17: *MAINTAIN Shelter PAR and CONTINUE assessment.* This step specifies to maintain current shelter PAR and continue to monitor and reassess impediments to evacuation of affected sectors during changing conditions.

    15.1 Additional sectors may have shelter recommended based on wind direction changes. In contrast, a shelter order that is not necessary prevents families from reuniting when it would be beneficial for evacuation readiness.

    15.2 The step re-directs flow path back to Step Par-14 to reassess if dose is below five times EPA limits of 1 rem TEDE or 5 rem CDE criteria. This loop of executing shelter PAR and reassessing for changes in impediments to evacuation and dose assessment continue until answer to Step Par-14 or Par-16 is YES, or entry to PAR flowchart is no longer met.

16. Step Par-18: Upon removal of impediments to evacuation, this step determines if plant condition or radiological conditions remain at a GE level or if likelihood of containment failure with significant source term from fuel damage exists that would exceed PAGs.

    16.1 *Plant and Rad conditions remain at GE* is asking if plant conditions or radiological conditions are still met therefore justifying requirement to change PAR to evacuation.

        16.1.1 IF GE plant and radiological conditions continue to be in effect, THEN evacuation PAR will be required.

        16.1.2 IF plant and radiological conditions have improved and GE no longer met,
        THEN evacuation not required. However, expanding existing shelter PAR may be required in Step Par-19.

    16.2 *Containment with source term > PAGs may fail* is asking:

        16.2.1 IF containment is threatened such that it may fail due to continuing degraded or adverse plant conditions (greater than 56 psig or greater than deflagration concentrations) with a release of source term that would result in exceeding EPA limits of 1 rem TEDE or 5 rem CDE,
        THEN expansion of existing PAR to additional areas or sectors where EPA limits of 1 rem TEDE or 5 rem CDE may be exceeded is warranted.

16.3 IF answer is YES,
THEN flow path directs to Step Par-13 to issue GE PAR which requires evacuation recommendation regardless of evacuation impediments or controlled releases due to anticipated dose.

16.4 IF answer is NO,
THEN flow path directs to Step Par-19 action to **EXPAND** current PAR only to areas where EPA limits are exceeded.

17. Step Par-19: **EXPAND** *PAR only to areas where 1 rem TEDE or 5 rem CDE could be exceeded.* The step directs PAR be expanded to new areas only if dose assessment is projected to exceed EPA PAGs in those areas.

17.1 This action differs from Step Par-9 which **EXPANDS** current PAR to new sectors based on either dose assessment results or plant conditions.

ATTACHMENT 5 AFFECTED SECTOR DETERMINATION (INFORMATION USE)

1. **DETERMINE** affected sectors using Attachment 5, Table 1, and following steps:

   1.1 **OBTAIN** wind speed and direction from Dose Assessment Report.

   1.2 IF Dose Assessment Report not available,
   THEN **OBTAIN** wind speed and direction from PMIS Display PMIS05 (primary) or National Weather Service at 402-359-4381 (backup).

      1.2.1 PMIS CNS-DOSE displays 15 minute average.

      1.2.2 IF National Weather Service used,
      THEN obtain 15 minute average data, if available, or 15 previous 1 minute readings.

      1.2.3 IF multiple release heights with different wind directions greater than EPA limits of 1 rem TEDE or 5 rem CDE,
      THEN both wind directions should be used to determine affected sectors.©[1]

   1.3 **ROUND** wind direction to nearest whole number.

NOTE 1 – Wind direction is given as "wind from" (easterly wind or wind direction 90° means wind blowing from east to west).

NOTE 2 – Effective Sector Determination table below adds additional sector if wind direction is within 3° of sector edge.

Table 1

WIND FROM (°)	SECTORS AFFECTED	WIND FROM (°)	SECTORS AFFECTED	WIND FROM (°)	SECTORS AFFECTED
351 to 9	H, J, K	126 to 144	P, Q, R	261 to 279	D, E, F
10 to 13	H, J, K, L	145 to 148	P, Q, R, A	280 to 283	D, E, F, G
14 to 31	J, K, L	149 to 166	Q, R, A	284 to 301	E, F, G
32 to 35	J, K, L, M	167 to 170	Q, R, A, B	302 to 305	E, F, G, H
36 to 54	K, L, M	171 to 189	R, A, B	306 to 324	F, G, H
55 to 58	K, L, M, N	190 to 193	R, A, B, C	325 to 328	F, G, H, J
59 to 76	L, M, N	194 to 211	A, B, C	329 to 346	G, H ,J
77 to 80	L, M, N, P	212 to 215	A, B, C, D	347 to 350	G, H, J, K
81 to 99	M, N, P	216 to 234	B, C, D		
100 to 103	M, N, P, Q	235 to 238	B, C, D, E	There is no O Sector	
104 to 121	N, P, Q	239 to 256	C, D, E		
122 to 125	N, P, Q, R	257 to 260	C, D, E, F	There is no I Sector	

1.4 IF wind shift observed following initial PAR issuance, THEN downwind sectors should be **RE-EVALUATED** for inclusion of new sectors (do not delete sectors) as follows:

NOTE 1 – During extreme wind shift due to severe weather, PARs should not be extended to sectors wind is rapidly shifting through unless shown by dose assessment projection EPA PAGs would be exceeded in those sectors for short duration of release.

NOTE 2 – Plant condition based PARs should be based on wind direction after extreme wind shift has subsided.

1.4.1 IF GE declared based on existing plant conditions, THEN PARs should be **EXPANDED** to include new sectors.

1.4.2 IF updated dose projections in new sectors exceed EPA limits of 1 rem TEDE or 5 rem CDE,
THEN PARs should be **EXPANDED** to include new sectors.

1.4.2.1 PARs should also be **EXPANDED** if containment may fail due to continuing degraded or adverse plant conditions (greater than 56 psig or greater than deflagration concentrations) and if containment source term would result in exceeding EPA limits of 1 rem TEDE or 5 rem CDE.

1.4.2.2 IF General Emergency would not be declared in new sectors based on existing plant conditions and updated dose projections in new sectors do not exceed EPA 400 PAGs,
THEN updated PARs are not required.

ATTACHMENT 6 EPA 400 INFORMATION (INFORMATION USE)

EPA 400 PROTECTIVE ACTION GUIDELINES (PAGs) FOR EARLY PHASE OF A NUCLEAR INCIDENT

PROTECTIVE ACTION	PROJECTED DOSE (PAG)	COMMENTS
Evacuation	1 rem TEDE rem [2,3] or 5 rem CDE (Thyroid)	Evacuation or, for some situations, sheltering[1] should normally be initiated at 1 rem. Further guidance is provided in Section 2.3.1 of EPA 400.

[1] Sheltering may be preferred protective action WHEN it will provide protection equal to or greater than evacuation, based on consideration of factors, such as, source term characteristics, and temporal or other site-specific conditions (refer to Section 2.3.1 of EPA 400).

[2] Sum of effective dose equivalent resulting from exposure to external sources and committed effective dose equivalent (CEDE) incurred from all significant inhalation pathways during early phase. Committed dose equivalents to thyroid and to skin may be 5 and 50 times larger, respectively.

[3] Evacuation will normally be recommended by NPPD for areas with projected TEDE dose of greater than 1 rem or CDE (Thyroid) dose of greater than 5 rem.

ATTACHMENT 7 COOPER NUCLEAR STATION 10 MILE EMERGENCY PREPAREDNESS ZONE MAP (INFORMATION USE)

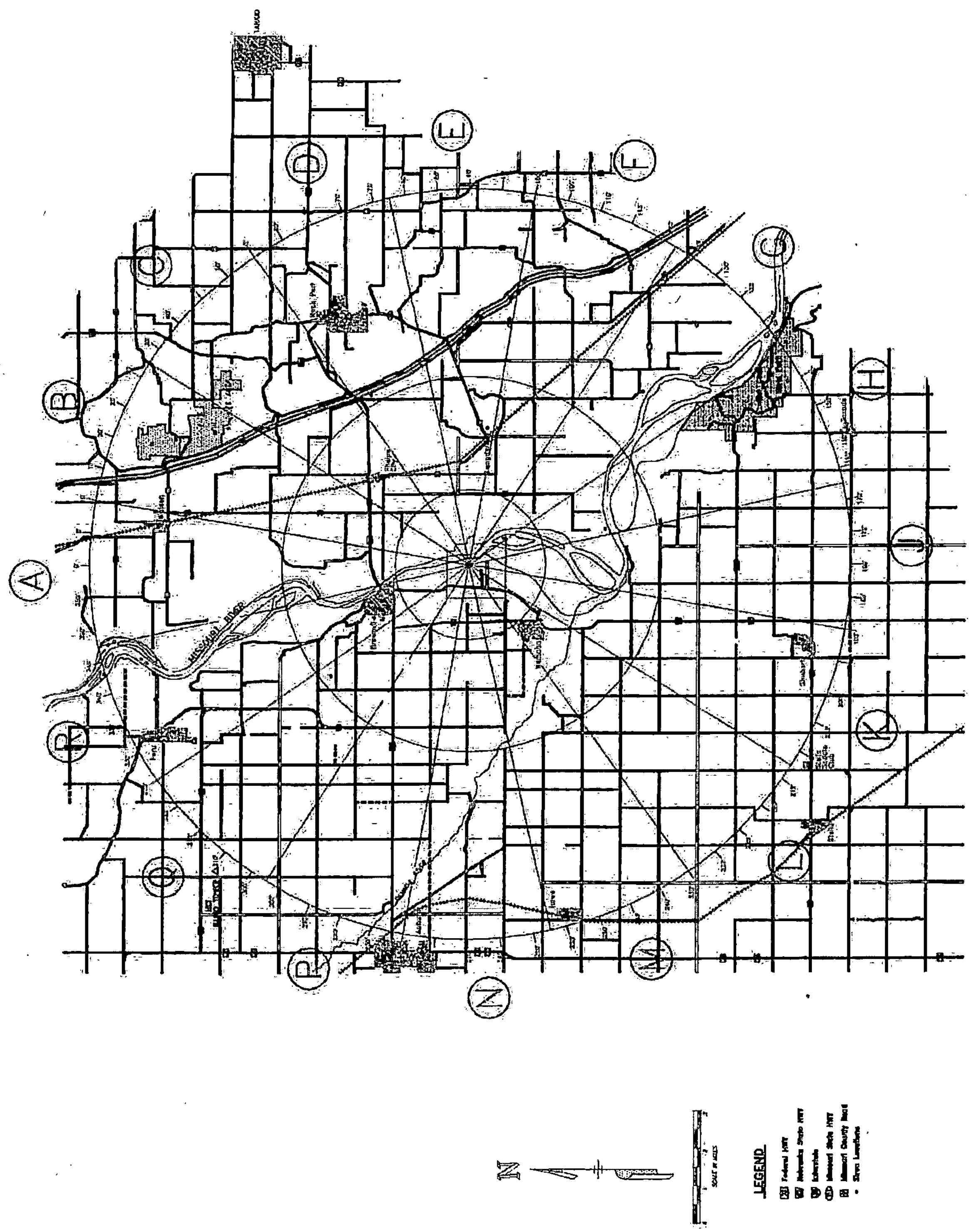


Figure 1

ATTACHMENT 8 INFORMATION SHEET (INFORMATION USE)

1. PURPOSE

1.1 Evaluates actual or projected dose or plant conditions to EPA Protective Action Guidelines (PAGs) to determine appropriate Protective Action Recommendations (PARs) to be made to County or State governments.

2. PRECAUTIONS AND LIMITATIONS

2.1 Conditions beyond the scope of this procedure may exist which, in the opinion of the Emergency Director, override the criteria contained in this procedure.

2.2 Plant conditions, off-site release of radioactive material, and impediments to evacuation are considered in the determination of PARs.

3. DISCUSSION

3.1 Dose estimates are calculated according to dose assessment methodology described in EPIPs 5.7.17 and 5.7.17.1. These dose estimates are referred to as projected dose. A protective action is an action taken to avoid or reduce future dose when benefits derived from such action are sufficient to offset any undesirable features of protective action.

3.2 A Rapidly Progressing Severe Accident (RPSA) is a General Emergency (GE) with rapid loss of containment integrity (emergency action levels indicate containment barrier loss) and loss of ability to cool core. This path is used for scenarios in which containment integrity can be determined as bypassed or immediately lost during GE with core damage.

3.3 PARs are required at a GE (modified by known impediments to evacuation and release characteristics).

4. PAR FLOWCHART AND AFFECTED SECTOR DETERMINATION HARD CARD

4.1 Information contained in Attachment 3, Protective Action Recommendation Flowchart, and Attachment 5, Affected Sector Determination, may be reformatted and placed on HARDCARDS similar to EOP Flowcharts. These PAR HARDCARDS will be controlled per this attachment. This information will be word for word but may be formatted differently using different font sizes or color backgrounds to assist visual presentation.

4.2 Attachment 3, Protective Action Recommendation Flowchart, is sized and formatted to be laminated and mounted on 11 by 17 inch foam board.

4.3 Each PAR HARDCARD will be labeled with a Revision Data box that will list latest revision. This data will match information below:

Table 1	
HARD CARD	REVISION
Attachment 3, Protective Action Recommendations Flowchart	Revision 07
Attachment 5, Affected Sector Determination Table 5-1	Revision 02

4.4 It is not necessary that HARDCARD revision number be revised with each revision of this procedure.

4.5 IF HARDCARD, Attachment 3 flowchart, or Attachment 5 table is revised, THEN Attachment 8 must be revised to reflect new PAR HARDCARD revision data.

4.6 PAR HARDCARD distribution will be made to following locations:

- Control Room.
- Simulator.
- Emergency Operations Facility.
- Technical Support Center.
- Emergency Preparedness Office.

5. RECORDS

5.1 No quality records are generated by this EPIP.

6. REFERENCES

6.1 CODES AND STANDARDS

6.1.1 Environmental Protection Agency EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, May 1992.

6.1.2 NEI 12-10, Revision 1, Guideline for Developing a Licensee Protective Action Recommendation Procedure Using NUREG-0654 Supplement 3.

6.1.3 NPPD Emergency Plan for CNS.

6.1.4 NRC RIS 2004-13, Consideration of Sheltering In Licensee's Range of Protective Action Recommendations.

6.1.5 NRC RIS 2005-08, Endorsement of Nuclear Energy Institute (NEI) Guidance "Range of Protective Actions for Nuclear Power Plant Incidents".

6.1.6 NRC Safety Evaluation Report (SER), TAC Number ME0849 - ADAMS Accession Number ML100080231.

6.1.7 Nuclear Energy Institute (NEI) Document 99-01, Revision 5, Methodology for Development of Emergency Action Levels.

6.1.8 NUREG BR-0150, Volume 1, Revision 4.

6.1.9 NUREG-0654/FEMA-REP-1, Revision 1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants.

6.1.10 NUREG-0654/FEMA-REP-1, Revision 1, Supplement 3, Criteria for Preparation and Evaluation of Radiological Response Plans and Preparedness in Support of Nuclear Power Plants, Guidance for Protective Action Strategies.

6.1.11 Reactor Safety Study, Appendix VI, WASH 1400, October 1975.

6.2 PROCEDURES

6.2.1 Emergency Plan Implementing Procedure 5.7.1, Emergency Classification.

6.2.2 Emergency Plan Implementing Procedure 5.7.6, Notification.

6.2.3 Emergency Plan Implementing Procedure 5.7.17, Dose Assessment.

6.2.4 Emergency Plan Implementing Procedure 5.7.17.1, Dose Assessment (Manual).

6.3 MISCELLANEOUS

6.3.1 Evacuation Time Estimates for Nebraska and Missouri.

6.3.2 RCR 2002-0181, Action 3. Notify State and Local Authorities within 15 minutes of making change to PAR.

6.4 NRC COMMITMENTS

6.4.1 ©[1] NRC Commitment NLS2013056-01, Capability to Perform Off-Site Dose Assessment during event involving multiple release sources. Commitment affects Attachment 5, Step 1.2.3.