HA6

ECL: ~~Alert~~ ALERT

Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations.

Operating Mode Applicability: ~~All~~ALL

~~Example~~ Emergency Action Level~~s~~:

(1) An event has resulted in plant control being transferred from the Control Room to ~~(site-specific remote shutdown panels and local control stations).~~ the Auxiliary Shutdown Panel (ASP).~~)~~.~~:~~

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.
Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Escalation of the ~~emergency classification level~~ EMERGENCY CLASSIFICATION ~~would~~LEVEL would be via IC HS6.

HA6: EAL-1 Selection Basis:

The Auxiliary Shutdown Panel (ASP) is identified in 0POP04-ZO-0001, Control Room Evacuation, as the location where plant control is transferred in the event of a Control Room evacuation.

REFERENCES:

1. Procedure 0POP04-ZO-0001, Rev. 35, Control Room Evacuation

~~**Developer Notes:**~~
~~The "site-specific remote shutdown panels and local control stations" are the panels and control stations referenced in plant procedures used to cooldown and shutdown the plant from a location(s) outside the Control Room.~~
~~ECL Assignment Attributes: 3.1.2.B~~

ECL: ~~Alert~~ ALERT

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

Operating Mode Applicability: ~~All~~**ALL**

~~**Example**~~ **Emergency Action Level~~s~~:**

(1) Other conditions exist which, in the judgment of the Emergency Director, indicate that 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~~ SECURITY EVENT that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. ~~Any~~ANY releases are expected to be limited to small fractions of the EPA ~~Protective Action Guideline~~ PROTECTIVE ACTION GUIDELINE exposure levels.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the ~~emergency classification level~~ EMERGENCY CLASSIFICATION ~~description~~LEVEL description for an ~~Alert~~ ALERT.

HA7: EAL-1 Selection Basis:

N/A

REFERENCE:

N/A

HS1

ECL: ~~Site Area Emergency~~ SITE AREA EMERGENCY

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA.

Operating Mode Applicability: ~~All~~ALL

~~Example~~ Emergency Action Level~~s~~:

(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by ~~the (site-specific security shift supervision). any~~ ANY of the following personnel in Table H1:

Table H1: Security Supervision
• Security Force Supervisor
• Acting Security Manager
• Security Manager

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the PROTECTED AREA. This event will require rapid response and assistance due to the possibility for damage to plant equipment. Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and ~~Independent Spent Fuel Storage Installation~~ INDEPENDENT SPENT FUEL STORAGE INSTALLATION Security Program]*.

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The ~~Site Area Emergency SITE~~SITE AREA EMERGENCY declaration will mobilize ORO resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

This IC does not apply to ~~a HOSTILE ACTION directed at an ISFSI PROTECTED AREA located outside the plant PROTECTED AREA; such an attack should be assessed using IC HA1. It also does not apply to~~ incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

Emergency plans and implementing procedures are public documents; therefore, EALs ~~should~~ do not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information ~~should be~~ is contained in ~~non-public documents such as~~ the Security Plan.

Escalation of the ~~emergency classification level~~ EMERGENCY CLASSIFICATION LEVEL would be via IC HG1.

HS1: EAL-1 Selection Basis:

The positions of Security Force Supervisor, Acting Security Manager, and Security Manager were included since any of these positions could be activated prior to meeting this EAL. The Security Force Supervisor is a 24-hour position, the Acting Security Manager is activated after an Unusual Event has been declared and the Security Manager is activated after an Alert is declared.

REFERENCES:

1. 0ERP01-ZV-SH03, Rev. 12, Acting Security Manager
2. 0ERP01-ZV-TS08, Rev. 16, Security Manager

~~**Developer Notes:**~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

~~Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.~~

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~See the related Developer Note in Appendix B, Definitions, for guidance on the development of a scheme definition for the PROTECTED AREA.~~

~~ECL Assignment Attributes: 3.1.3.D~~

HS6

ECL: ~~Site Area Emergency SITE~~SITE AREA EMERGENCY

Initiating Condition: Inability to control a key safety function from outside the Control Room.

Operating Mode Applicability: ~~All~~**ALL**

~~**Example**~~ **Emergency Action Level~~s~~:**

Note: The Emergency Director should declare the ~~Site Area Emergency SITE~~SITE AREA EMERGENCY promptly upon determining that ~~(site-specific number of minutes)~~ 15 minutes has been exceeded, or will likely be exceeded.

(1) a. An event has resulted in plant control being transferred from the Control Room to ~~(site-specific remote shutdown panels and local control stations).~~the Auxiliary Shutdown Panel (ASP).~~the following locations:~~

AND

b. Control of **ANY** of the following key safety functions in Table H2 is not reestablished within ~~(site-specific number of minutes)~~ 15 minutes.~~).~~

- ~~Reactivity control~~
- ~~Core cooling [*PWR*] / RPV water level [*BWR*]~~
- ~~RCS heat removal~~

Table H2: Safety Functions
• Reactivity control
• Core cooling
• RCS heat removal

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not "control" is established at the ~~remote safe shutdown location(s)~~ Auxiliary Shutdown Panel is based on Emergency Director judgment. The Emergency Director is expected to make a reasonable, informed judgment within ~~(the site-specific time for transfer)~~ 15~~10~~ minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

Escalation of the ~~emergency classification level~~ EMERGENCY CLASSIFICATION LEVEL would be via IC FG1 or CG1.

HS6: EAL-1 Selection Basis:

The Auxiliary Shutdown Panel (ASP) is identified in 0POP04-ZO-0001, Control Room Evacuation, as the location where plant control is transferred in the event of a Control Room evacuation. The 15 minute timeframe to control the key safety functions is identified as site specific information

REFERENCE:

Procedure 0POP04-ZO-0001, Rev. 35, Control Room Evacuation

1.
2.
3. ~~**Developer Notes:**~~
4. ~~The "site-specific remote shutdown panels and local control stations" are the panels and control stations referenced in plant procedures used to cooldown and shutdown the plant from a location(s) outside the Control Room.~~
5. ~~The "site-specific number of minutes" is the time in which plant control must be (or is expected to be) reestablished at an alternate location as described in the site-specific fire response analyses. Absent a basis in the site-specific analyses, 15 minutes should be used. Another time period may be used with appropriate basis/justification.~~
6. ~~ECL Assignment Attributes: 3.1.3.B~~

HS7

ECL: ~~Site Area Emergency SITE~~SITE AREA EMERGENCY

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a ~~Site Area Emergency SITE~~SITE AREA EMERGENCY.

Operating Mode Applicability: ~~All~~**ALL**

~~Example~~ Emergency Action Level~~s~~:

(1) Other conditions exist which in the judgment of the Emergency Director indicate that 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~~ANY releases are not expected to result in exposure levels which exceed EPA ~~Protective Action Guideline~~ PROTECTIVE ACTION GUIDELINE exposure levels beyond the ~~site boundary~~ SITE BOUNDARY.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the ~~emergency classification level~~ EMERGENCY CLASSIFICATION LEVEL description for a ~~Site Area Emergency~~ SITE AREA EMERGENCY.

HS7: EAL-1 Selection Basis:

N/A

REFERENCE:

N/A

ECL: ~~General Emergency GENERAL~~GENERAL EMERGENCY

Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the ~~facility~~ FACILITY.

Operating Mode Applicability: ~~All~~ALL

~~Example~~ Emergency Action Level~~s~~:

(1) a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by ~~**the (site-specific security shift supervision).**~~ ~~**any**~~ **ANY** of the following in Table H1:

Table H1: Security Supervision
• Security Force Supervisor
• Acting Security Manager
• Security Manager

AND

b. **EITHER** of the following has occurred:

1. **ANY** of the following safety functions in Table H2 cannot be controlled or maintained.

- ~~Reactivity control~~
- ~~Core cooling [*PWR*] / RPV water level [*BWR*]~~
- ~~RCS heat removal~~

Table H2: Safety Functions
• Reactivity control
• Core cooling
• RCS heat removal

OR

2. Damage to spent fuel has occurred or is IMMINENT.

Basis:

This IC addresses an event in which a HOSTILE FORCE has taken physical control of the ~~facility~~ FACILITY to the extent that the plant staff can no longer operate equipment necessary to maintain key safety functions. It also addresses a HOSTILE ACTION leading to a loss of physical control that results in actual or IMMINENT damage to spent fuel due to 1) damage to a spent fuel pool cooling system (e.g., pumps, heat exchangers, controls, etc.) or, 2) loss of spent fuel pool integrity such that sufficient water level cannot be maintained.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and ~~Independent Spent Fuel Storage Installation~~ INDEPENDENT SPENT FUEL STORAGE INSTALLATION Security Program]*.

Emergency plans and implementing procedures are public documents; therefore, EALs ~~should~~ do not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information ~~should is be~~ is contained in ~~non-public documents such as~~ the Security Plan.

HG1: EAL-1 Selection Basis:

The positions of Security Force Supervisor, Acting Security Manager, and Security Manager were also included since any of these positions could be activated prior to meeting this EAL.

REFERENCES:

1. 0ERP01-ZV-SH03, Rev. 12, Acting Security Manager
2. 0ERP01-ZV-TS08, Rev. 16, Security Manager

~~**Developer Notes:**~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

~~Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.~~

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~See the related Developer Note in Appendix B, Definitions, for guidance on the development of a scheme definition for the PROTECTED AREA.~~

~~ECL Assignment Attributes: 3.1.4.D~~

HG7

ECL: ~~General Emergency~~ GENERAL EMERGENCY

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

Operating Mode Applicability: ~~**All**~~**ALL**

~~**Example**~~ **Emergency Action Level~~s~~:**

(1) Other conditions exist which in the judgment of the Emergency Director indicate that 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~~FACILITY. Releases can be reasonably expected to exceed EPA ~~Protective Action Guideline~~ PROTECTIVE ACTION GUIDELINE exposure levels offsite for more than the immediate site area.

Basis:

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

HG7: EAL-1 Selection Basis:

N/A

REFERENCE:

N/A

11 SYSTEM MALFUNCTION ICS/EALS

Table S-1: Recognition Category "S" Initiating Condition Matrix

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
SU1 Loss of ~~all~~**ALL** offsite AC power capability to emergency buses for 15 minutes or longer. *Op. Modes: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1,2,3,4*	**SA1** Loss of ~~all~~**ALL** but one AC power source to emergency buses for 15 minutes or longer. *Op. Modes: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1,2,3,4*	**SS1** Loss of ~~all~~**ALL** offsite and ~~all~~**ALL** onsite AC power to emergency buses for 15 minutes or longer. *Op. Modes: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1,2,3,4*	**SG1** Prolonged loss of ~~all~~**ALL** offsite and ~~all~~**ALL** onsite AC power to emergency buses. *Op. Modes: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1,2,3,4*
SU2 UNPLANNED loss of Control Room indications for 15 minutes or longer. *Op. Modes: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1,2,3,4*	**SA2** UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress. *Op. Modes: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1,2,3,4*		
SU3 Reactor coolant activity greater than Technical Specification allowable limits. *Op. Modes: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1,2,3,4*			
SU4 RCS leakage for 15 minutes or longer. *Op. Modes: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1,2,3,4*			
SU5 Automatic or manual ~~(~~trip ~~[PWR] / scram [BWR])~~ fails to shutdown the reactor. *Op. Modes: ~~Power Operation~~1,2 ~~1~~*	**SA5** Automatic or manual ~~(~~trip ~~[PWR] / scram [BWR])~~ fails to shutdown the reactor, and subsequent manual actions taken at the reactor control ~~consoles~~ panels are not successful in shutting down the reactor. *Op. Modes: ~~Power Operation~~ 1,2*	**SS5** Inability to shutdown the reactor causing a challenge to ~~(~~core cooling ~~[PWR] / RPV water level [BWR])~~ or RCS heat removal. *Op. Modes: ~~Power Operation~~ 1,2*	

Table S-1: Recognition Category "S" Initiating Condition Matrix (cont.)

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
SU6 Loss of ~~all~~ALL onsite or offsite communications capabilities. *Op. Modes:* ~~*Power Operation, Startup, Hot Standby, Hot Shutdown*~~ *1,2,3,4*			
SU7 Failure to isolate containment or loss of containment pressure control. ~~*[PWR]*~~ ~~*Op. Modes: Power Operation, Startup, Hot Standby, Hot Shutdown*~~ *1,2,3,4*			
		SS8 Loss of ~~all~~ALL Vital DC power for 15 minutes or longer. *Op. Modes:* ~~*Power Operation, Startup, Hot Standby, Hot Shutdown*~~ *1,2,3,4*	**SG8** Loss of ~~all~~ALL AC and Vital DC power sources for 15 minutes or longer. *Op. Modes:* ~~*Power Operation, Startup, Hot Standby, Hot Shutdown*~~ *1,2,3,4*
	SA9 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. *Op. Modes:* ~~*Power Operation, Startup, Hot Standby, Hot Shutdown*~~ *1,2,3,4*		

SU1

ECL: ~~Notification of Unusual Event~~ UNUSUAL EVENT

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

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1, 2, 3, 4

~~**Example**~~ **Emergency Action Level~~s~~:**

Note: The Emergency Director should declare the ~~Unusual Event~~ UNUSUAL EVENT promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) Loss of ~~**ALL**~~**ALL** offsite AC power capability to ~~**(site-specific emergency buses)**~~ ~~**all**~~**ALL** three 4160V AC ESF Buses~~Busses~~ for 15 minutes or longer.

Basis:

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC emergency buses. This condition represents a potential reduction in the level of safety of the plant.

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the emergency buses, whether or not the buses are powered from it.

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

Escalation of the ~~emergency classification level~~ EMERGENCY CLASSIFICATION LEVEL would be via IC SA1.

SU1: EAL-1 Selection Basis:

N/A

REFERENCES:

1. 0POP04-AE-0001, Rev. 44, First Response to Loss of Any or All 13.8 KV or 4.16 KV Bus
2. 0POP04-AE-0004, Rev. 15, Loss of Power to One or More 4.16 KV ESF Bus
3. 0PSP03-EA-0002, Rev. 32, ESF Power Availability
4. Drawing 00000E0AAAA, Rev. 24, Single Line Diagram, Main One Line Diagram, Unit No. 1 & 2

~~**Developer Notes:**~~
~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~
~~At multi-unit stations, the EALs may credit compensatory measures that are proceduralized and can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, "swing"~~

~~generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to supply offsite AC power to an affected unit via a cross-tie to a companion unit may credit this power source in the EAL provided that the planned cross-tie strategy meets the requirements of 10 CFR 50.63.~~

~~ECL Assignment Attributes: 3.1.1.A~~

SU2

ECL: ~~Notification of Unusual Event~~ UNUSUAL EVENT

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1, 2, 3, 4

~~Example~~ **Emergency Action Level~~s~~:**

Note: The Emergency Director should declare the UNUSUAL EVENT~~Unusual Event~~ promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) An UNPLANNED event results in the inability to monitor one or more of the following parameters in Table S1 from within the Control Room for 15 minutes or longer.

~~[BWR parameter list]~~	~~[PWR parameter list]~~
~~Reactor Power~~	~~Reactor Power~~
~~RPV Water Level~~	~~RCS Level~~
~~RPV Pressure~~	~~RCS Pressure~~
~~Primary Containment Pressure~~	~~In-Core/Core Exit Temperature~~
~~Suppression Pool Level~~	~~Levels in at least (site-specific number) two steam generators~~
~~Suppression Pool Temperature~~	~~Steam Generator Auxiliary or Emergency Feed Water Flow~~

Table S1: Plant Parameters
• Reactor Power • RCS Level • RCS Pressure • Core Exit Temperature • Levels in at least two steam generators • Steam Generator Auxiliary Feed Water Flow

Basis:

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the

Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling ~~[PWR] / RPV level [BWR]~~ and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for ~~reactor vessel level~~RCS level ~~[PWR] / RPV water level [BWR]~~ cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication. Escalation of the ~~emergency classification level~~ EMERGENCY CLASSIFICATION LEVEL would be via IC SA2.

SU2: EAL-1 Selection Basis:

The parameters listed were from NEI 99-01, Rev. 6 with the exception of steam generators. Two steam generators is a site-specific parameter for the minimum number of steam generators needed for plant cooldown and shutdown.

REFERENCES:

1. 0POP05-EO-E020, Rev. 11, Faulted Steam Generator Isolation
2. 0POP05-EO-FRH1, Rev. 23, Response to Loss of Secondary Heat Sink

~~**Developer Notes:**~~

~~In the PWR parameter list column, the "site-specific number" should reflect the minimum number of steam generators necessary for plant cooldown and shutdown. This criterion may also specify whether the level value should be wide-range, narrow-range or both, depending upon the monitoring requirements in emergency operating procedures.~~

~~Developers may specify either pressurizer or reactor vessel level in the PWR parameter column entry for RCS Level.~~

~~The number, type, location and layout of Control Room indications, and the range of possible failure modes, can challenge the ability of an operator to accurately determine, within the time period available for emergency classification assessments, if a specific percentage of indications have been lost. The approach used in this EAL facilitates prompt and accurate emergency classification assessments by focusing on the indications for a selected subset of parameters.~~

~~By focusing on the availability of the specified parameter values, instead of the sources of those values, the EAL recognizes and accommodates the wide variety of indications in nuclear power plant Control Rooms. Indication types and sources may be analog or digital, safety-related or not, primary or alternate, individual meter value or computer group display, etc.~~

~~A loss of plant annunciators will be evaluated for reportability in accordance with 10 CFR 50.72 (and the associated guidance in NUREG-1022), and reported if it significantly impairs the capability to perform emergency assessments. Compensatory measures for a loss of annunciation can be readily implemented and may include~~ increase~~d monitoring of main control boards and more frequent plant rounds by non-licensed operators. Their alerting function notwithstanding, annunciators do not provide the parameter values or specific component status information used to operate the plant, or process through AOPs or EOPs. Based on these considerations, a loss of annunciation is considered to be adequately addressed by reportability criteria, and therefore not included in this IC and EAL.~~

~~With respect to establishing event severity, the response to a loss of radiation monitoring data (e.g., process or effluent monitor values) is considered to be adequately bounded by the requirements of 10 CFR 50.72 (and associated guidance in NUREG-1022). The reporting of this event will ensure adequate plant staff and NRC awareness, and drive the establishment of appropriate compensatory measures and corrective actions. In addition, a loss of radiation monitoring data, by itself, is not a precursor to a more significant event.~~

~~Personnel at sites that have a Failure Modes and Effects Analysis (FMEA) included within the design basis of a digital I&C system should consider the FMEA information when developing their site-specific EALs.~~

~~Due to changes in the configurations of SAFETY SYSTEMS, including associated instrumentation and indications, during the cold shutdown, refueling, and defueled modes, no analogous IC is included for these modes of operation.~~

~~ECL Assignment Attributes: 3.1.1.A~~

SU3

ECL: ~~Notification of Unusual Event~~ UNUSUAL EVENT

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1, 2, 3, 4

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

(1) ~~(Site-specific radiation monitor)~~RT-8039 reading~~readings~~ greater than 30 µCi/cm^3~~(site-specific value)~~.

(2) Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications.

- Greater than 1µCi~~1uCi~~/gm Dose Equivalent I-131
- Greater than 100/ Ē~~E~~ bar µCi /gm gross activity

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the ~~emergency classification level~~ EMERGENCY CLASSIFICATION LEVEL would be via ICs FA1 or the Recognition Category ~~A~~R ICs.

SU3: EAL-1 Selection Basis:

RT-8039 is the Failed Fuel radiation monitor and samples via the CVCS letdown line. The value 30 µCi/cm^3 is the reading that is equivalent to 1µCi/gm Dose Equivalent I-131. The monitor value in this EAL is the calculated monitor response if the RCS activity were equivalent to 1µCi/gm Dose Equivalent I-131. The value is based on Calculation STPNOC013-CALC-003. The value used in this EAL was conservatively truncated by approximately 5% to ensure the value is readily assessable.

SU3: EAL-2 Selection Basis:

The Technical Specification limits for RCS activity is greater than 1µCi/gm Dose Equivalent I-131 or greater than 100/Ē bar µCi /gm gross activity.

REFERENCES:

1. Calculation No. -STPNOC013-CALC-003, Gross Failed Fuel Monitor Response to Rise RCS Activity (RT-8039 EAL Threshold)~~XXX~~
2. STP Technical Specification Section 3/4.4.8 Specific Activity.

~~**Developer Notes:**~~

~~For EAL #1 – Enter the radiation monitor(s) that may be used to readily identify when RCS activity levels exceed Technical Specification allowable limits. This EAL may be developed using different methods and sites should use existing capabilities to address it (e.g., development of new capabilities is not required). Examples of existing methods/capabilities include:~~

- ~~An installed radiation monitor on the letdown system or air ejector.~~
- ~~A hand-held monitor or deployed detector reading with pre-calculated conversion values or readily implementable conversion calculation capability.~~

~~The monitor reading values should correspond to an RCS activity level approximately at Technical Specification allowable limits.~~

~~If there is no existing method/capability for determining this EAL, then it should not be included. IC evaluation will be based on EAL #2.~~

~~For EAL#2 – Developers may reword the EAL to include the reactor coolant activity parameter(s) specified in Technical Specifications and the associated allowable limit(s) (e.g., values for dose equivalent I-131 and gross activity, time-dependent or transient values, etc.). If this approach is selected, all RCS activity allowable limits should be included.~~

~~ECL Assignment Attributes: 3.1.1.A and 3.1.1.B~~

SU4

ECL: ~~Notification of Unusual Event~~ UNUSUAL EVENT

Initiating Condition: RCS leakage for 15 minutes or longer.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1, 2, 3, 4

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

Note: The Emergency Director should declare the UNUSUAL EVENT~~Unusual Event~~ promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) RCS unidentified or pressure boundary leakage greater than ~~(site-specific value)~~ 10 gpm for 15 minutes or longer.

(2) RCS identified leakage greater than ~~(site-specific value)~~ 25 gpm for 15 minutes or longer.

(3) Leakage from the RCS ~~to a location outside containment~~ greater than 150 gallons per day through ~~**ANY**~~**ANY** one steam generator.~~25 gpm for 15 minutes or longer.~~

Basis:

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

EAL #1 and EAL #2 are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications). EAL #3 addresses a RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These EALs thus apply to leakage into the containment, a secondary-side system (e.g., steam generator tube leakage~~ in a PWR~~) or a location outside of containment.

The leak rate values for each EAL were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). EAL #1 uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. ~~For PWRs, a~~ AnaAn emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated).~~ For BWRs, a stuck-open Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.~~

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

Escalation of the ~~emergency classification level~~ EMERGENCY CLASSIFICATION LEVEL would be via ICs of Recognition Category ~~A~~R or F.

SU4: EAL-1 Selection Basis:

The STP Technical Specifications limit for unidentified leakage from the RCS is 1 gpm. NEI 99-01 Rev. 6 states to use the higher of the Technical Specification limit or 10 gpm.

SU4: EAL-2 Selection Basis:

The STP Technical Specifications limit for identified leakage from the RCS is 10 gpm. NEI 99-01 Rev. 6 requirements are to use the higher of the Technical Specification limit or 25 gpm.

SU4: EAL-3 Selection Basis:

The STP Technical Specifications limit for primary-to-secondary leakage is 150 gallons per day through any one steam generator.

REFERENCES:

1. STP Technical Specification Section 3.4.6.2 Reactor Coolant System Operational Leakage.

~~**Developer Notes:**~~
~~EAL #1 For the site-specific leak rate value, enter the higher of 10 gpm or the value specified in the site's Technical Specifications for this type of leakage.~~
~~EAL #2 For the site-specific leak rate value, enter the higher of 25 gpm or the value specified in the site's Technical Specifications for this type of leakage.~~
~~For sites that have Technical Specifications that do not specify a leakage type for steam generator tube leakage, developers should include an EAL for tube leakage greater than 25 gpm for 15 minutes or longer.~~
~~ECL Assignment Attributes: 3.1.1.A~~

ECL: ~~Notification of Unusual Event~~ UNUSUAL EVENT

Initiating Condition: Automatic or manual ~~(~~trip ~~[PWR] / scram [BWR])~~ fails to shutdown the reactor.

Operating Mode Applicability: ~~Power Operation~~ 1, 2

~~Note: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.~~

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

Note: A manual action is **ANY** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

(1) a. An automatic ~~(~~trip ~~[PWR] / scram [BWR])~~ did not shutdown the reactor.

AND

b. A subsequent manual action taken at the reactor control ~~consoles~~panels is successful in shutting down the reactor.

(2) a. A manual trip ~~(trip [PWR] / scram [BWR])~~ did not shutdown the reactor.

AND

b. **EITHER** of the following:

1. A subsequent manual action taken at the reactor control ~~consoles~~panels is successful in shutting down the reactor.

OR

~~3.~~ 2. A subsequent automatic ~~(~~trip ~~[PWR] / scram [BWR])~~ is successful in shutting down the reactor.

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(~~trip ~~[PWR] / scram [BWR])~~ that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control ~~consoles~~panels or an automatic ~~(~~trip ~~[PWR] / scram [BWR])~~ is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor ~~(~~trip~~ [PWR] / scram [BWR])~~, operators will promptly initiate manual actions at the reactor control ~~consoles~~panels to shutdown the reactor (e.g., initiate a manual reactor ~~(~~trip~~ [PWR] / scram [BWR])~~). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor ~~(~~trip~~ [PWR] / scram [BWR])~~ is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control ~~consoles~~panels to shut down the reactor (e.g., initiate a manual reactor ~~(~~trip~~ [PWR] / scram [BWR])~~) using a different switch). Depending upon several factors, the initial or subsequent effort to manually ~~(~~trip~~ [PWR] / scram [BWR])~~ the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor ~~(~~trip ~~[PWR] / scram [BWR])~~signal. If a subsequent manual or automatic ~~(~~trip~~ [PWR] / scram [BWR])~~ is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control ~~consoles~~panels is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual ~~(~~trip~~ [PWR] / scram [BWR])~~). This action does not include manually driving in control rods or implementation of boron injection strategies. ~~Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consolespanels". Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action. [BWR]~~
The plant response to the failure of an automatic or manual reactor ~~(~~trip~~ [PWR] / scram [BWR])~~ will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control ~~consoles~~panels are also unsuccessful in shutting down the reactor, then the ~~emergency classification level~~ EMERGENCY CLASSIFICATION LEVEL will escalate to an ~~Alert~~ ALERT via IC SA5. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA5 or FA1, an ~~Unusual Event~~ UNUSUAL EVENT declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor ~~(~~trip~~ [PWR] / scram [BWR])~~ signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic ~~(~~trip~~ [PWR] / scram [BWR])~~ and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the ~~(~~trip~~ [PWR] / scram [BWR])~~ failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

SU5: EAL-1, EAL-2 Selection Basis:

N/A

REFERENCES:

1. 0POP03-ZG-0004, Rev. 45, Reactor Startup
2. 0POP03-ZG-0005, Rev. 86, Plant Startup to 100%

~~**Developer Notes:**~~

~~This IC is applicable in any Mode in which the actual reactor power level could exceed the power level at which the reactor is considered shutdown. A PWR with a shutdown reactor power level that is less than or equal to the reactor power level which defines the lower bound of Power Operation (Mode 1) will need to include Startup (Mode 2) in the Operating Mode Applicability. For example, if the reactor is considered to be shutdown at 3% and Power 159 Operation starts at >5%, then the IC is also applicable in Startup Mode.~~

~~Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level).~~

~~The term "reactor control consoles" may be replaced with the appropriate site-specific term (e.g., main control boards).~~

~~ECL Assignment Attributes: 3.1.1.A~~

SU6

ECL: ~~Notification of Unusual Event~~ UNUSUAL EVENT

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

Operating Mode Applicability: ~~*Power Operation, Startup, Hot Standby, Hot Shutdown*~~ 1, 2, 3, 4

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

(1) Loss of **ALL** of the following onsite communication methods listed in Table S2.

~~(site-specific list of communications methods)~~

(2) Loss of **ALL** of the following Offsite Response Organization (ORO)~~ORO~~ communications methods listed in Table S2.

~~(site-specific list of communications methods)~~

(3) Loss of **ALL** of the following NRC communications methods listed in Table S2.

~~(site-specific list of communications methods)ENS line~~

Table S2: Communications Methods

METHOD	EAL-1 ONSITE	EAL-2 ORO	EAL-3 NRC
• Plant PA system	X		
• Plant Radios	X		
• Plant telephone system	X	X	X
• Satellite phones		X	X
• Direct line from Control Rooms to Bay City		X	X
• Microwave Lines to Houston		X	X
• Security radio to Matagorda County		X	
• Dedicated Ring-down lines		X	
• ENS line			X

Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

EAL #1- addresses a total loss of the communications methods used in support of routine plant operations.

EAL #2- addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are ~~(see Developer Notes)~~Matagorda County Sheriff's Office, and Texas Department of Public Safety Disaster District in Pierce.

EAL #3- addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

SU6: EAL-1, EAL-2, EAL-3 Selection Basis:

Lines not included for offsite communications to ORO and NRC included links that would need relaying of information. Links were obtained from procedures 0PGP05-ZV-0011, Emergency Communications.

REFERENCES:

1. 0PGP05-ZV-0011, Emergency Communications

~~**Developer Notes:**~~
~~EAL #1 - The "site-specific list of communications methods" should include all communications methods used for routine plant communications (e.g., commercial or site telephones, page-party systems, radios, etc.). This listing should include installed plant equipment and components, and not items owned and maintained by individuals.~~
~~161 EAL #2 - The "site-specific list of communications methods" should include all communications methods used to perform initial emergency notifications to OROs as described in the site Emergency Plan. The listing should include installed plant equipment and components, and not items owned and maintained by individuals. Example methods are ring-down/dedicated telephone lines, commercial telephone lines, radios, satellite telephones and internet-based communications technology.~~
~~In the Basis section, insert the site-specific listing of the OROs requiring notification of an emergency declaration from the Control Room in accordance with the site Emergency Plan, and typically within 15 minutes.~~
~~EAL #3 - The "site-specific list of communications methods" should include all communications methods used to perform initial emergency notifications to the NRC as described in the site Emergency Plan. The listing should include installed plant equipment and components, and not items owned and maintained by individuals. These methods are typically the dedicated Emergency Notification System (ENS) telephone line and commercial telephone lines.~~
~~ECL Assignment Attributes: 3.1.1.C~~

ECL: ~~Notification of Unusual Event~~ UNUSUAL EVENT

Initiating Condition: Failure to isolate containment or loss of containment pressure control. ~~[PWR]~~

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1, 2, 3, 4

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

(1) a. Failure of containment to isolate when required by an actuation signal.

AND

b. **ALL** required penetrations are not isolated~~closed~~ within 15 minutes of the actuation signal.

(2) a. Containment pressure greater than ~~(site-specific pressure)~~9.5 psig.

AND

b. ~~Less than one full train of (site-specific system or equipment)~~ No Containment Spray~~containment spray~~ train is operating per design for 15 minutes or longer.

Basis:

This IC addresses a failure of one or more containment penetrations to automatically isolate ~~(close)~~ when required by an actuation signal. It also addresses an event that results in high containment pressure with a concurrent failure of containment pressure control systems. Absent challenges to another fission product barrier, either condition represents potential degradation of the level of safety of the plant.

~~For~~ EAL #1~~,~~ the containment isolation signal must be generated as the result on an off-normal/accident condition (e.g., a safety injection or high containment pressure); a failure resulting from testing or maintenance does not warrant classification. The determination of containment and penetration status – isolated or not isolated – should be made in accordance with the appropriate criteria contained in the plant AOPs and EOPs. The 15-minute criterion is included to allow operators time to manually isolate the required penetrations, if possible.

EAL #2- addresses a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment spray~~s or ice condenser fans~~) are either lost or performing in a degraded manner.

~~163~~ This event would escalate to a ~~Site Area Emergency~~ SITE AREA EMERGENCY in accordance with IC FS1 if there were a concurrent loss or potential loss of either the Fuel Clad or RCS fission product barriers.

SU7: EAL-1 Selection Basis:

N/A

SU7: EAL-2 Selection Basis:

If containment pressure reaches 9.5 psig, Containment Spray will actuate. If no train of Containment Spray is operating per design, the ability to lower containment pressure is compromised. One train of Containment Spray (Technical Specifications 3/4.6.2) is defined as one containment spray system capable of taking a suction from the RWST and transferring suction to the containment sump.

REFERENCES:

1. 0POP05-EO-F005, Rev. 1, Containment Critical Safety Function Status Tree
2. 0POP05-EO-FRZ1, Rev. 9, Response to High Containment Pressure
3. Technical Specifications 3/4.6.2

~~2.~~

~~3. **Developer Notes:**~~

~~4. Enter the "site-specific pressure" value that actuates containment pressure control systems (e.g., containment spray). Also enter the site-specific containment pressure control system/equipment that should be operating per design if the containment pressure actuation setpoint is reached. If desired, specific condition indications such as parameter values can also be entered (e.g., a containment spray flow rate less than a certain value).~~

~~5. EAL #2 is not applicable to the U.S. Evolutionary Power Reactor (EPR) design.~~

~~6. ECL Assignment Attributes: 3.1.1.A~~

ECL: ~~Alert~~ ALERT

Initiating Condition: Loss of ~~all~~ALL but one AC power source to emergency buses for 15 minutes or longer.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1, 2, 3, 4

~~Example~~ Emergency Action Level~~s~~:

Note: The Emergency Director should declare the ~~Alert~~ ALERT promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) a. AC power capability to **~~(site-specific emergency buses)~~ ~~all~~ALL** three 4160V~~4160v~~ AC ESF Buses is reduced to a single power source for 15 minutes or longer.

AND

b. **~~ANY~~ANY** additional single power source failure will result in a loss of **~~all~~ALL** AC power to SAFETY SYSTEMS.

Basis:

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an emergency bus. Some examples of this condition are presented below

.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being ~~back-~~fed from the unit main generator.
- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being ~~back-~~fed from an onsite or offsite power source.

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

Escalation of the ~~emergency classification level~~ EMERGENCY CLASSIFICATION LEVEL would be via IC SS1.

SA1: EAL-1 Selection Basis:

This EAL is similar to IC CU2, except this EAL applies only to Modes 1-4.

REFERENCES:

1. 0POP04-AE-0001, Rev. 44, First Response to Loss of Any or All 13.8 KV or 4.16 KV Bus
2. 0POP04-AE-0004, Rev. 15, Loss of Power to One or More 4.16 KV ESF Bus
3. 0PSP03-EA-0002, Rev. 32, ESF Power Availability
4. Drawing 00000E0AAAA, Rev. 24, Single Line Diagram, Main One Line Diagram, Unit No. 1 & 2

~~**Developer Notes:**~~

~~For a power source that has multiple generators, the EAL and/or Basis section should reflect the minimum number of operating generators necessary for that source to provide required power to an AC emergency bus. For example, if a backup power source is comprised of two generators (i.e., two 50% capacity generators sized to feed 1 AC emergency bus), the EAL and Basis section must specify that both generators for that source are operating.~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~Developers should modify the bulleted examples provided in the basis section, above, as needed to reflect their site-specific plant designs and capabilities.~~

~~The EALs and Basis should reflect that each independent offsite power circuit constitutes a single power source. For example, three independent 345kV offsite power circuits (i.e., incoming power lines) comprise three separate power sources. Independence may be determined from a review of the site-specific UFSAR, SBO analysis or related loss of electrical power studies.~~

~~The EAL and/or Basis section may specify use of a non-safety-related power source provided that operation of this source is recognized in AOPs and EOPs, or beyond design basis accident response guidelines (e.g., FLEX support guidelines). Such power sources should generally meet the "Alternate ac source" definition provided in 10 CFR 50.2.~~

~~At multi-unit stations, the EALs may credit compensatory measures that are proceduralized and can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, "swing" generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to supply offsite AC power to an affected unit via a cross-tie to a companion unit may credit this power source in the EAL provided that the planned cross-tie strategy meets the requirements of 10 CFR 50.63.~~

~~ECL Assignment Attributes: 3.1.2.B~~

ECL: ~~Alert~~ ALERT

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1, 2, 3, 4

~~Example~~ Emergency Action Levels:

Note: The Emergency Director should declare the ~~Alert~~ ALERT promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) a. An UNPLANNED event results in the inability to monitor one or more of the following parameters in Table S1 from within the Control Room for 15 minutes or longer.

~~[*BWR parameter list*]~~	~~[*PWR parameter list*]~~
~~Reactor Power~~	~~Reactor Power~~
~~RPV Water Level~~	~~RCS Level~~
~~RPV Pressure~~	~~RCS Pressure~~
~~Primary Containment Pressure~~	~~In-Core/Core Exit Temperature~~
~~Suppression Pool Level~~	~~Levels in at least (site-specific number) two steam generators~~
~~Suppression Pool Temperature~~	~~Steam Generator Auxiliary or Emergency Feed Water Flow~~

Table S1: Plant Parameters
• Reactor Power • RCS Level • RCS Pressure • Core Exit Temperature • Levels in at least two steam generators • Steam Generator Auxiliary Feed Water Flow

AND

b. **ANY** of the following transient events in progress.

- Automatic or manual runback greater than 25% thermal reactor power
- Electrical load rejection greater than 25% full electrical load
- Reactor ~~scram [BWR] /~~ trip ~~[PWR]~~

- ECCS (SI) actuation

~~Thermal power oscillations greater than 10% [BWR]~~

Basis:

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling ~~[*PWR*] / RPV level [*BWR*]~~ and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for ~~reactor vessel level~~RCS level ~~[*PWR*] / RPV water level [*BWR*]~~ cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the ~~emergency classification level~~ EMERGENCY CLASSIFICATION LEVEL would be via ICs FS1 or IC ~~AS1~~ RS1.

SA2: EAL-1 Selection Criteria:

The plant parameters listed are from NEI 99-01, Rev. 6. Two steam generators were selected as a site-specific parameter for the minimum number of steam generators needed for plant cooldown and shutdown.

REFERENCES:

1. 0POP05-EO-EO20, Rev. 11, Faulted Steam Generator Isolation
2. 0POP05-EO-FRH1, Rev. 23, Response to Loss of Secondary Heat Sink

~~**Developer Notes:**~~

~~In the PWR parameter list column, the "site-specific number" should reflect the minimum number of steam generators necessary for plant cooldown and shutdown. This criterion may also specify whether the level value should be wide range, narrow range or both, depending upon the monitoring requirements in emergency operating procedures.~~

~~Developers may specify either pressurizer or reactor vessel level in the PWR parameter column entry for RCS Level.~~

~~Developers should consider if the "transient events" list needs to be modified to better reflect site-specific plant operating characteristics and expected responses.~~

~~The number, type, location and layout of Control Room indications, and the range of possible failure modes, can challenge the ability of an operator to accurately determine, within the time period available for emergency classification assessments, if a specific percentage of indications have been lost. The approach used in this EAL facilitates prompt and accurate emergency classification assessments by focusing on the indications for a selected subset of parameters.~~

~~By focusing on the availability of the specified parameter values, instead of the sources of those values, the EAL recognizes and accommodates the wide variety of indications in nuclear power plant Control Rooms. Indication types and sources may be analog or digital, safety-related or not, primary or alternate, individual meter value or computer group display, etc.~~

~~168 A loss of plant annunciators will be evaluated for reportability in accordance with 10 CFR 50.72 (and the associated guidance in NUREG-1022), and reported if it significantly impairs the capability to perform emergency assessments. Compensatory measures for a loss of annunciation can be readily implemented and may include increased monitoring of main control boards and more frequent plant rounds by non-licensed operators. Their alerting function notwithstanding, annunciators do not provide the parameter values or specific component status information used to operate the plant, or process through AOPs or EOPs. Based on these considerations, a loss of annunciation is considered to be adequately addressed by reportability criteria, and therefore not included in this IC and EAL.~~

~~With respect to establishing event severity, the response to a loss of radiation monitoring data (e.g., process or effluent monitor values) is considered to be adequately bounded by the requirements of 10 CFR 50.72 (and associated guidance in NUREG-1022). The reporting of this event will ensure adequate plant staff and NRC awareness, and drive the establishment of appropriate compensatory measures and corrective actions. In addition, a loss of radiation monitoring data, by itself, is not a precursor to a more significant event.~~

~~Personnel at sites that have a Failure Modes and Effects Analysis (FMEA) included within the design basis of a digital I&C system should consider the FMEA information when developing their site-specific EALs.~~

~~Due to changes in the configurations of SAFETY SYSTEMS, including associated instrumentation and indications, during the cold shutdown, refueling, and defueled modes, no analogous IC is included for these modes of operation.~~

~~ECL Assignment Attributes: 3.1.2.B~~

ECL: ~~Alert~~ ALERT

Initiating Condition: Automatic or manual ~~(~~trip ~~[PWR] / scram [BWR])~~ fails to shutdown the reactor, and subsequent manual actions taken at the reactor control~~consoles~~ panels~~control consoles~~ are not successful in shutting down the reactor.

Operating Mode Applicability: ~~Power Operation~~ 1, 2

~~**Note:** A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.~~

~~**Example**~~ **Emergency Action Level~~s~~:**

Note: A manual action is **ANY** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

(1) a. An automatic or manual ~~(~~trip ~~[PWR] / scram [BWR])~~ did not shutdown the reactor.

AND

b. Manual actions taken at the reactor control ~~consoles~~panels are not successful in shutting down the reactor.

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(~~trip ~~[PWR] / scram [BWR])~~ that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control ~~consoles~~panels to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control ~~consoles~~panels since this event entails a significant failure of the RPS.

A manual action at the reactor control ~~consoles~~panels is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor ~~(~~trip ~~[PWR] / scram [BWR])~~). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control ~~consoles~~panels (e.g., locally opening breakers). ~~Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".~~

~~Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action. [*BWR*]~~
The plant response to the failure of an automatic or manual reactor ~~(~~trip ~~[PWR] / scram [BWR])~~ will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the

failure to shutdown the reactor is prolonged enough to cause a challenge to the core cooling ~~[PWR] / RPV water level [BWR]~~ or RCS heat removal safety functions, the ~~emergency classification level~~ EMERGENCY CLASSIFICATION LEVEL will escalate to a ~~Site Area Emergency~~ SITE AREA EMERGENCY via IC SS5. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. ~~Absent the plant conditions needed to meet either IC SS5 or FS1, an Alert declaration is appropriate for this event.~~

It is recognized that plant responses or symptoms may also require an ~~Alert~~ ALERT declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

SA5: EAL-1 Selection Basis:

N/A

REFERENCES:

1. 0POP05-E0-FRS1, Rev. 17, Response to Nuclear Power Generation - ATWS

~~**Developer Notes:**~~
~~This IC is applicable in any Mode in which the actual reactor power level could exceed the power level at which the reactor is considered shutdown. A PWR with a shutdown reactor power level that is less than or equal to the reactor power level which defines the lower bound of Power Operation (Mode 1) will need to include Startup (Mode 2) in the Operating Mode Applicability. For example, if the reactor is considered to be shutdown at 3% and Power Operation starts at >5%, then the IC is also applicable in Startup Mode. Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level).~~
~~The term "reactor control consoles" may be replaced with the appropriate site-specific term (e.g., main control boards).~~
~~ECL Assignment Attributes: 3.1.2.B~~

SA9

ECL: ~~Alert~~ ALERT

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1, 2, 3, 4

~~Example~~ Emergency Action Level~~s~~:

(1) a. The occurrence of **ANY** of the following hazardous events listed in Table S3:

~~Seismic event (earthquake)~~
~~Internal or external flooding event~~
~~High winds or tornado strike~~
~~FIRE~~
~~EXPLOSION~~
~~(site-specific hazards) Predicted or actual breach of Main Cooling Reservoir retaining dike along North Wall.~~
~~Other events with similar hazard characteristics as determined by the Shift Manager~~

Table S3: Hazardous Events
• Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Predicted or actual breach of Main Cooling Reservoir retaining dike along North Wall. • Other events with similar hazard characteristics as determined by the Shift Manager

AND

b. **EITHER** of the following:

1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode.

   OR

2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.

Basis:

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

EAL# 1.b.1- addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

EAL# 1.b.2- addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the ~~emergency classification level~~ EMERGENCY CLASSIFICATION LEVEL would be via IC FS1 or ~~A~~RS1~~AS1~~.

SA9: EAL-1 Selection Basis:

The listed hazards are from NEI 99-01 Rev.6 with the exception of the Main Cooling Reservoir breach along the north wall which was included because it is a credible hazard and analyzed in the STPEGS UFSAR.

REFERENCES:

1. STPEGS UFSAR, Section 3.4.1, Flood Protection

~~**Developer Notes:**~~
~~For (site-specific hazards), developers should consider including other significant, site-specific hazards to the bulleted list contained in EAL 1.a (e.g., a seiche).~~
~~Nuclear power plant SAFETY SYSTEMS are comprised of two or more separate and redundant trains of equipment in accordance with site-specific design criteria.~~
~~ECL Assignment Attributes: 3.1.2.B~~

ECL: ~~Site Area Emergency~~ SITE AREA EMERGENCY

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

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1, 2, 3, 4

~~**Example**~~ **Emergency Action Level~~s~~:**

Note: The Emergency Director should declare the ~~Site Area Emergency SITE~~SITE AREA EMERGENCY promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) Loss of **ALL** offsite and **ALL** onsite AC power to ~~**(site-specific emergency buses) all**~~**ALL** three 4160V AC ESF Buses for 15 minutes or longer.

Basis:

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

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

Escalation of the ~~emergency classification level~~ EMERGENCY CLASSIFICATION LEVEL would be via ICs ~~AG1~~RG1, FG1 or SG1.

SS1: EAL-1 Selection Criteria:

N/A

~~**SS1: EAL-1**~~ **REFERENCES:**

1. 0POP04-AE-0001, Rev. 44, First Response to Loss of Any or All 13.8 KV or 4.16 KV Bus
2. 0POP04-AE-0004, Rev. 15, Loss of Power to One or More 4.16 KV ESF Bus
3. 0PSP03-EA-0002, Rev. 32, ESF Power Availability
4. Drawing 00000E0AAAA, Rev. 24, Single Line Diagram, Main One Line Diagram, Unit No. 1 & 2

~~**Developer Notes:**~~

~~For a power source that has multiple generators, the EAL and/or Basis section should reflect the minimum number of operating generators necessary for that source to provide adequate power to an AC emergency bus. For example, if a backup power source is comprised of two generators (i.e., two 50% capacity generators sized to feed 1 AC emergency bus), the EAL and Basis section must specify that both generators for that source are operating.~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~The EAL and/or Basis section may specify use of a non-safety-related power source provided that operation of this source is controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines). Such power sources should generally meet the "Alternate ac source" definition provided in 10 CFR 50.2.~~

~~At multi-unit stations, the EALs may credit compensatory measures that are proceduralized and can be implemented within 15 minutes. Consider capabilities such as power source cross-ties, "swing" generators, other power sources described in abnormal or emergency operating procedures, etc. Plants that have a proceduralized capability to supply offsite AC power to an affected unit via a cross-tie to a companion unit may credit this power source in the EAL provided that the planned cross-tie strategy meets the requirements of 10 CFR 50.63.~~

~~ECL Assignment Attributes: 3.1.3.B~~

ECL: ~~Site Area Emergency SITE~~ SITE AREA EMERGENCY

Initiating Condition: Inability to shutdown the reactor causing a challenge to ~~(~~core cooling ~~[PWR] / RPV water level [BWR])~~ or RCS heat removal.

Operating Mode Applicability: ~~Power Operation~~ 1, 2

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

(1) a. An automatic or manual ~~(~~trip ~~[PWR] / scram [BWR])~~ did not shutdown the reactor.

AND

b. ~~**All**~~**ALL** manual actions to shutdown the reactor have been unsuccessful.

AND

c. **EITHER** of the following conditions exists:

- ~~(Site-specific indication of an inability to adequately remove heat from the core)~~ Core Cooling – Red entry conditions~~condition~~ met
- ~~(Site-specific indication of an inability to adequately remove heat from the RCS)~~ Heat Sink-Red entry conditions~~condition~~ met

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(~~trip ~~[PWR] / scram [BWR])~~that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a ~~Site Area Emergency SITE~~SITE AREA EMERGENCY.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shutdown the reactor. The inclusion of this IC and EAL ensures the timely declaration of a ~~Site Area Emergency SITE~~SITE AREA EMERGENCY in response to prolonged failure to shutdown the reactor.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Escalation of the ~~emergency classification level~~ EMERGENCY CLASSIFICATION LEVEL would be via IC ~~A~~RG1~~AG1~~RG1 or FG1.

SS5: EAL-1 Selection Basis:

Core Cooling - Red entry conditions met (CETs > 1200° F) is the site specific indication of the inability to adequately remove heat from the core. Heat Sink - Red entry conditions met (NR level in All SG < 14% [34%] AND total AFW flow to SG < 576 GPM) is the site specific indication of the inability to remove heat from the RCS.

~~SS1: EAL-1~~ REFERENCES:

1. Procedure 0POP05-EO-F002, Rev. 2, Core Cooling Critical Safety Function Status Tree
2. Procedure 0POP05-EO-F003, Rev. 6, Heat Sink Critical Safety Function Status Tree

~~**Developer Notes:**~~

~~This IC is applicable in any Mode in which the actual reactor power level could exceed the power level at which the reactor is considered shutdown. A PWR with a shutdown reactor power level that is less than or equal to the reactor power level which defines the lower bound of Power Operation (Mode 1) will need to include Startup (Mode 2) in the Operating Mode Applicability. For example, if the reactor is considered to be shutdown at 3% and Power Operation starts at >5%, then the IC is also applicable in Startup Mode. Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level).~~

~~Site-specific indication of an inability to adequately remove heat from the core:~~

~~*[BWR]* – Reactor vessel water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level (as described in the EOP bases).~~

~~*[PWR]* – Insert site-specific values for an incore/core exit thermocouple temperature and/or reactor vessel water level that drives entry into a core cooling restoration procedure (or otherwise requires implementation of prompt restoration actions). Alternately, a site may use incore/core exit thermocouple temperatures greater than 1,200oF and/or a reactor vessel water level that corresponds to approximately the middle of active fuel. Plants with reactor vessel level instrumentation that cannot measure down to approximately the middle of active fuel should use the lowest on-scale reading that is not above the top of active fuel. If the lowest on-scale reading is above the top of active fuel, then a reactor vessel level value should not be included.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters used in the Core Cooling Red Path.~~

~~Site-specific indication of an inability to adequately remove heat from the RCS:~~

~~*[BWR]* – Use the Heat Capacity Temperature Limit. This addresses the inability to remove heat via the main condenser and the suppression pool due to high pool water temperature.~~

~~*[PWR]* – Insert site-specific parameters associated with inadequate RCS heat removal via the steam generators. These parameters should be identical to those used for the Inadequate Heat Removal threshold Fuel Clad Barrier Potential Loss 2.B and threshold RCS Barrier Potential Loss 2.A in the PWR EAL Fission Product Barrier Table.~~

~~ECL Assignment Attributes: 3.1.3.B~~

ECL: ~~Site Area Emergency~~ -SITE AREA EMERGENCY

Initiating Condition: Loss of ~~all~~ALL Vital DC power for 15 minutes or longer.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1, 2, 3, 4

~~Example~~ Emergency Action Level~~s~~:

Note: The Emergency Director should declare the ~~Site Area Emergency~~ SITE AREA EMERGENCY promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) Indicated voltage is less than ~~(site-specific bus voltage value)~~ 105.5 VDC~~volts~~ on **ALL** ~~(site-specific Vital DC busses)~~ Class 1E 125 VDC battery buses for 15 minutes or longer.

Basis:

This IC addresses a loss of Vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

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

Escalation of the ~~emergency classification level~~ EMERGENCY CLASSIFICATION LEVEL would be via ICs ~~AG1~~RG1, FG1 or SG8.

SS8: EAL-1 Selection Basis:

Minimum voltage for Class 1E 125 VDC battery buses was determined in calculation 13-DJ-006 Rev.3 and determined to be 105.5 volts. At 105.5 volts or less, 0POP05-E0-EC00, Loss of All AC Power directs the operators to open the battery output breakers.

REFERENCES:

1. 0POP05-E0-EC00, Rev. 23, Loss of All AC Power

~~Developer Notes:~~

~~The "site-specific bus voltage value" should be based on the minimum bus voltage necessary for adequate operation of SAFETY SYSTEM equipment. This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is usually near the minimum voltage selected when battery sizing is performed.~~

~~The typical value for an entire battery set is approximately 105 VDC. For a 60 cell string of batteries, the cell voltage is approximately 1.75 Volts per cell. For a 58 string battery set, the minimum voltage is approximately 1.81 Volts per cell.~~

~~The "site-specific Vital DC busses" are the DC busses that provide monitoring and control capabilities for SAFETY SYSTEMS.~~

~~ECL Assignment Attributes: 3.1.3.B~~

ECL: ~~General Emergency GENERAL~~GENERAL EMERGENCY

Initiating Condition: Prolonged loss of ~~all~~ALL offsite and ~~all~~ALL onsite AC power to emergency buses.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1, 2, 3, 4

~~Example~~ **Emergency Action Level~~s~~:**

Note: The Emergency Director should declare the ~~General Emergency GENERAL~~GENERAL EMERGENCY promptly upon determining that ~~(site-specific hours)~~ 4 hours has been exceeded, or will likely be exceeded.

(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to ~~(site-specific emergency buses)~~~~**all**~~**ALL** three 4160V AC ESF Buses.

AND

b. **EITHER** of the following:

- Restoration of at least one ~~AC emergency~~ 4160VAC ESF bus in less than ~~(site-specific hours)~~ 4 hours is not likely.
- ~~(Site-specific indication of an inability to adequately remove heat from the core)~~ Core Cooling – Red entry condition met

Basis:

This IC addresses a prolonged loss of all power sources to AC emergency buses. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A prolonged loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

The EAL should require declaration of a ~~General Emergency GENERAL~~GENERAL EMERGENCY prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from ~~Site Area Emergency SITE~~SITE AREA EMERGENCY will occur if it is projected that power cannot be restored to at least one AC emergency bus by the end of four (4) hours.~~the analyzed station blackout coping period.~~ Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is a~~n~~ ~~increased~~higher likelihood of challenges to multiple fission product barriers.

The estimate for restoring at least one emergency bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, protective actions for the public.

The EAL will also require a ~~General Emergency~~GENERAL EMERGENCY declaration if the loss of AC power results in parameters that indicate an inability to adequately remove decay heat from the core.

SG1: EAL-1 Selection Basis:

The prolonged loss of all onsite and all offsite AC power coupled with Core Cooling - Red entry conditions (CETs > 1200° F) are sufficient indications of the inability to remove heat from the core.

Station Blackout does not include the loss of available AC power to buses fed by station batteries through inverters, or by Alternate AC (AAC) sources as defined in NUMARC 87-00. The STPEGS Station Blackout position credits any one of the three Standby Diesel Generators as the AAC source. The required coping duration category determined for STPEGS Station Blackout is a minimum of four hours, based on the guidance of NUMARC 87-00, Section 3. STPEGS meets this requirement and forms the basis for the four hour time period.

REFERENCES:

1. 0POP04-AE-0001, Rev. 44, First Response to Loss of Any or All 13.8 KV or 4.16 KV Bus
2. 0POP04-AE-0004, Rev. 15, Loss of Power to One of More 4.16 KV ESF Buses
3. 0PSP03-EA-0002, Rev. 32, ESF Power Availability
4. Drawing 00000E0AAAA, Rev. 24, Single Line Diagram, Main One Line Diagram, Unit No. 1 & 2
5. 0POP05-EO-F002, Rev. 2, Core Cooling Critical Safety Function Status Tree
6. 0POP05-EO-EC00, Rev. 23, Loss of All AC Power
7. STPEGS UFSAR Section 8.3.4, Station Blackout

~~**Developer Notes:**~~

~~Although this IC and EAL may be viewed as redundant to the Fission Product Barrier ICs, it is included to provide for a more timely escalation of the emergency classification level.~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~The "site-specific hours" to restore AC power to an emergency bus should be based on the station blackout coping analysis performed in accordance with 10 CFR § 50.63 and Regulatory Guide 1.155, *Station Blackout*.~~

~~Site-specific indication of an inability to adequately remove heat from the core:~~

~~[*BWR*] Reactor vessel water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level (as described in the EOP bases).~~

~~[*PWR*] Insert site-specific values for an incore/core exit thermocouple temperature and/or reactor vessel water level that drive entry into a core cooling restoration procedure (or otherwise requires implementation of prompt restoration actions). Alternately, a site may use incore/core exit thermocouple temperatures greater than 1,200oF and/or a reactor vessel water level that corresponds to approximately~~

~~the middle of active fuel. Plants with reactor vessel level instrumentation that cannot measure down to approximately the middle of active fuel should use the lowest on-scale reading that is not above the top of active fuel. If the lowest on-scale reading is above the top of active fuel, then a reactor vessel level value should not be included.~~

~~For plants that have implemented Westinghouse Owners Group Emergency Response Guidelines, enter the parameters used in the Core Cooling Red Path.~~

~~ECL Assignment Attributes: 3.1.4.B~~

ECL: ~~General Emergency~~ GENERAL EMERGENCY

Initiating Condition: Loss of ~~all~~ALL AC and Vital DC power sources for 15 minutes or longer.

Operating Mode Applicability: ~~Power Operation, Startup, Hot Standby, Hot Shutdown~~ 1, 2, 3, 4

Emergency Action Levels:

Note: The Emergency Director should declare the ~~General Emergency~~ GENERAL EMERGENCY promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to ~~**(site-specific emergency buses)**~~ ~~**all**~~**ALL** three 4160V~~4160 V~~ AC ESF buses for 15 minutes or longer.

AND

b. Indicated voltage is less than ~~(site-specific bus voltage value)~~ 105.5 VDC~~volts DC~~ on ~~**ALL**~~**ALL** ~~(site-specific Vital DC busses)~~ Class 1E 125 VDC battery buses for 15 minutes or longer.

Basis:

This IC addresses a concurrent and prolonged loss of both AC and Vital DC power. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of Vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both AC and DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

SG8: EAL-1 Selection Basis:

This IC and EAL were included to address the operating experience for the March, 2011 accident at Fukushima Daiichi. Minimum voltage for Class 1E 125 VDC battery buses was determined in calculation 13-DJ-006 Rev.3 and determined to be 105.5 volts. At 105.5 volts or less, 0POP05-E0-EC00, Loss of All AC Power directs the operators to open the battery output breakers.

REFERENCES:

1. 0POP04-AE-0001, Rev. 44, First Response to Loss of Any or All 13.8 KV or 4.16 KV Bus
2. 0POP04-AE-0004, Rev. 15, Loss of Power to One of More 4.16 KV ESF Buses
3. 0PSP03-EA-0002, Rev. 32, ESF Power Availability
4. 0POP05-E0-EC00, Rev. 23, Loss of All AC Power
5. Drawing 00000E0AAAA, Rev. 24, Single Line Diagram, Main One Line Diagram, Unit No. 1 & 2

~~**Developer Notes:**~~

~~The "site-specific emergency buses" are the buses fed by offsite or emergency AC power sources that supply power to the electrical distribution system that powers SAFETY SYSTEMS. There is typically 1 emergency bus per train of SAFETY SYSTEMS.~~

~~The "site-specific bus voltage value" should be based on the minimum bus voltage necessary for adequate operation of SAFETY SYSTEM equipment. This voltage value should incorporate a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is usually near the minimum voltage selected when battery sizing is performed.~~

~~The typical value for an entire battery set is approximately 105 VDC. For a 60 cell string of batteries, the cell voltage is approximately 1.75 Volts per cell. For a 58 string battery set, the minimum voltage is approximately 1.81 Volts per cell.~~

~~The "site-specific Vital DC busses" are the DC busses that provide monitoring and control capabilities for SAFETY SYSTEMS.~~

~~This IC and EAL were added to Revision 6 to address operating experience from the March, 2011 accident at Fukushima Daiichi.~~

~~ECL Assignment Attributes: 3.1.4.B~~

APPENDIX A – ACRONYMS AND ABBREVIATIONS

AC .... Alternating Current
AOP .... Abnormal Operating Procedure
~~APRM .... Average Power Range Meter~~
ATWS .... Anticipated Transient Without Scram
~~B&W .... Babcock and Wilcox~~
~~BIIT .... Boron Injection Initiation Temperature~~
~~BWR .... Boiling Water Reactor~~
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
CTMT/CNMT .... Containment
CSF .... Critical Safety Function
CSFST .... Critical Safety Function Status Tree
DBA .... Design Basis Accident
DC .... Direct Current
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
EPG .... Emergency Procedure Guideline
~~EPIP .... Emergency Plan Implementing Procedure~~
~~EPR .... Evolutionary Power Reactor~~
~~EPRI .... Electric Power Research Institute~~
ERG .... Emergency Response Guideline
FEMA .... Federal Emergency Management Agency
FSAR .... Final Safety Analysis Report
GE .... ~~General Emergency~~GENERAL EMERGENCY
~~HCTL .... Heat Capacity Temperature Limit~~
~~HPCI .... High Pressure Coolant Injection~~
~~HSI .... Human System Interface~~
IC .... Initiating Condition
ID .... Inside Diameter
~~IPEEE .... Individual Plant Examination of External Events (Generic Letter 88-20)~~
ISFSI .... Independent Spent Fuel Storage Installation
Keff .... Effective Neutron Multiplication Factor
LCO .... Limiting Condition of Operation
LOCA .... Loss of Coolant Accident
~~MCR .... Main Control Room~~
MSIV .... Main Steam Isolation Valve
MSL .... Main Steam Line
mR, mRem, mrem, mREM .... milli-Roentgen Equivalent Man
MW .... Megawatt
NEI .... Nuclear Energy Institute
NPP .... Nuclear Power Plant

NRC .... Nuclear Regulatory Commission
NSSS .... Nuclear Steam Supply System
NORAD .... North American Aerospace Defense Command
(NO)UE .... (Notification Of) Unusual Event
NUMARC~~NUMARC9~~ .... Nuclear Management and Resources Council
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM~~/ODAM~~ .... Offsite Dose Calculation ~~(Assessment)~~ Manual
ORO .... Off-site Response Organization
PA .... Protected Area
~~PACS .... Priority Actuation and Control System~~
PAG .... Protective Action Guideline
~~PICS .... Process Information and Control System~~
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR .... Pressurized Water Reactor
~~PS .... Protection System~~
PSIG .... Pounds per Square Inch Gauge
R .... Roentgen
~~RCC .... Reactor Control Console~~
~~RCIC .... Reactor Core Isolation Cooling~~
RCS .... Reactor Coolant System
Rem, rem, REM .... Roentgen Equivalent Man
~~RETS .... Radiological Effluent Technical Specifications~~
RPS .... Reactor Protection System
RPV .... Reactor Pressure Vessel
RVWL~~IS~~ .... Reactor Vessel Water Level ~~Instrumentation System~~
~~RWCU .... Reactor Water Cleanup~~
SAR .... Safety Analysis Report
~~SAS .... Safety Automation System~~
~~SBO .... Station Blackout~~
SCBA .... Self-Contained Breathing Apparatus
SG .... Steam Generator
SI .... Safety Injection
~~SICS .... Safety Information and Control System~~
SPDS .... Safety Parameter Display System
~~SRO .... Senior Reactor Operator~~
TEDE .... Total Effective Dose Equivalent
TOAF .... Top of Active Fuel
TSC .... Technical Support Center
WOG .... Westinghouse Owners Group

APPENDIX B – DEFINITIONS

The following definitions are taken from Title 10, Code of Federal Regulations, and related regulatory guidance documents.

~~Alert~~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.

~~General Emergency~~GENERAL EMERGENCY: Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity 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 offsite for more than the immediate site area.

~~Notification of Unusual Event~~UNUSUAL EVENT ~~(NOUE~~UE~~)10~~: 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 offsite response or monitoring are expected unless further degradation of safety systems occurs.

~~Site Area Emergency~~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.
The following are key terms necessary for overall understanding the ~~NEI 99-01~~ emergency classification scheme.

~~Emergency Action Level~~ EMERGENCY ACTION LEVEL (EAL): A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given ~~emergency classification level.~~EMERGENCY CLASSIFICATION LEVEL.

~~Emergency Classification Level~~ EMERGENCY CLASSIFICATION LEVEL (ECL): One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The ~~emergency classification levels~~EMERGENCY CLASSIFICATION LEVELS, in ascending order of severity, are:

- ~~Notification of Unusual Event~~UNUSUAL EVENT ~~(NOUE~~ UE~~)~~
- ~~Alert~~ ALERT
- ~~Site Area Emergency~~ SITE AREA EMERGENCY (SAE)
- ~~General Emergency~~ GENERAL EMERGENCY (GE)
-

~~Fission Product Barrier Threshold~~FISSION PRODUCT BARRIER THRESHOLD: A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.
~~Initiating Condition~~
INITIATING CONDITION (IC): An event or condition that aligns with the definition of one of the four ~~emergency classification levels~~EMERGENCY CLASSIFICATION LEVELS by virtue of the potential or actual effects or consequences.

Selected terms used in ~~Initiating Condition~~ INITIATING CONDITION and EMERGENCY ACTION LEVEL ~~Emergency Action Level~~

EMERGENCY ACTION LEVEL statements are set in all capital letters (e.g., ALL CAPS). These words are defined terms that have specific meanings as used in this document. The definitions of these terms are provided below.

CONFINEMENT BOUNDARY: ~~(Insert a site-specific definition for this term.) **Developer Note**~~ The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.

CONTAINMENT CLOSURE: ~~(Insert a site-specific definition for this term.) **Developer Note**~~ The procedurally defined conditions or actions taken to secure ~~containment (primary or secondary for BWR)~~ and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

CREDIBLE SECURITY THREAT: Information received from a source determined to be reliable (e.g., law enforcement, government agency, etc.) or has been verified to be true or considered credible when: (1) Physical evidence supporting the threat exists, (2) Information independent from the actual threat message exists that supports the threat, or (3) A specific known group or organization claims responsibility for the threat.

EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.

FACILITY: The area and buildings within the PROTECTED AREA and the switchyard.

FAULTED: The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized~~. **Developer Note** – This term is applicable to PWRs only.~~

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.

HATCH MONITOR: Temporary monitor installed when Containment High Range Radiation Monitors RT-8050 and RT-8051 are out of service.

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

HOSTILE ACTION: An act toward a NPP 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 the NPP. 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).

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: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

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 LEVELS: As applied to radiological IC/EALs, the highest reading in the past twenty-four hours excluding the current peak value.

OWNER CONTROLLED AREA: ~~(Insert a site-specific definition for this term.) **Developer Note** – This term is typically taken to mean the site property owned by, or otherwise under the control of, the licensee. In some cases, it may be appropriate for a licensee to define a smaller area with a perimeter closer to the plant Protected Area perimeter (e.g., a site with a large OCA where some portions of the boundary may be a significant distance from the Protected Area). In these cases, developers should consider using the boundary defined by the Restricted or Secured Owner Controlled Area (ROCA/SOCA). The area and boundary selected for scheme use must be consistent with the description of the same area and boundary contained in the Security Plan.~~ The area surrounding the PROTECTED AREA where STP Nuclear Operating Company (STPNOC) reserves the right to restrict access, search personnel, and vehicles.

PROJECTILE: An object directed toward a NPP that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTIVE ACTION GUIDES (PAG): Environmental Protection Agency (EPA) guides for protective actions to safeguard against radiation exposure from nuclear incidents.

PROTECTED AREA: ~~(Insert a site-specific definition for this term.) **Developer Note** – This term is typically taken to mean the~~ The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

REFUELING PATHWAY ~~: (Insert a site-specific definition for this term.) **Developer Note** – This description should include~~ Includes all the cavities, tubes, canals and pools through which irradiated fuel may be moved, but not including the reactor vessel.

RUPTURE(D): The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection. ~~**Developer Note** – This term is applicable to PWRs only.~~

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related~~. **Developer Note** – This term may be modified to include the attributes of "safety-related" in accordance with 10 CFR 50.2 or other site-specific terminology, if desired.~~

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.

SECURITY EVENT: Any incident representing an attempted, threatened, of actual breach of the security system of reduction of the operational effectiveness of that system. A security event can result in either a SECURITY CONDITION or HOSTILE ACTION.

SITE BOUNDARY: The edge of the plant property whose access may be controlled by STPEGS. This boundary is congruent with the Exclusion Area Boundary for the purpose of offsite dose assessment.

UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

THYROID CDE: The dose equivalent to the thyroid from an intake of radioactive material by an individual during the 50-year period following the intake.

VALID: An indication, report or condition is considered to be VALID when it is verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy. This may be accomplished through an instrument channel check, response on related or redundant indicators, or direct observation by plant personnel. The verification methods should be completed in a manner the supports timely emergency declaration.

VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

~~APPENDIX C – PERMANENTLY DEFUELED STATION ICs/EALs~~

~~Recognition Category PD provides a stand-alone set of ICs/EALs for a Permanently Defueled nuclear power plant to consider for use in developing a site-specific emergency classification scheme. For development, it was assumed that the plant had operated under a 10 CFR § 50 license and that the operating company has permanently ceased plant operations. Further, the company intends to store the spent fuel within the plant for some period of time.~~

~~When in a permanently defueled condition, the plant licensee typically receives approval from the NRC for exemption from specific emergency planning requirements. These exemptions reflect the lowered radiological source term and risks associated with spent fuel pool storage relative to reactor at-power operation. Source terms and accident analyses associated with plausible accidents are documented in the station's Final Safety Analysis Report (FSAR), as updated. As a result, each licensee will need to develop a site-specific emergency classification scheme using the NRC-approved exemptions, revised source terms, and revised accident analyses as documented in the station's FSAR.~~

~~Recognition Category PD uses the same ECLs as operating reactors; however, the source term and accident analyses typically limit the ECLs to an Unusual Event and Alert. The Unusual Event ICs provide for an increased awareness of abnormal conditions while the Alert ICs are specific to actual or potential impacts to spent fuel. The source terms and release motive forces associated with a permanently defueled plant would not be sufficient to require declaration of a Site Area Emergency or General Emergency.~~

~~A permanently defueled station is essentially a spent fuel storage facility with the spent fuel is stored in a pool of water that serves as both a cooling medium (i.e., removal of decay heat) and shield from direct radiation. These primary functions of the spent fuel storage pool are the focus of the Recognition Category PD ICs and EALs. Radiological effluent IC and EALs were included to provide a basis for classifying events that cannot be readily classified based on an observable events or plant conditions alone.~~

~~Appropriate ICs and EALs from Recognition Categories A, C, F, H, and S were modified and included in Recognition Category PD to address a spectrum of the events that may affect a spent fuel pool. The Recognition Category PD ICs and EALs reflect the relevant guidance in Section 3 of this document (e.g., the importance of avoiding both over-classification and under-classification). Nonetheless, each licensee will need to develop their emergency classification scheme using the NRC-approved exemptions, and the source terms and accident analyses specific to the licensee. Security-related events will also need to be considered.~~

~~Table PD-1: Recognition Category "PD" Initiating Condition Matrix~~

~~UNUSUAL EVENT~~	~~ALERT~~
~~**PD-AU1** Release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.~~ ~~*Op. Modes: Not Applicable*~~	~~**PD-AA1** Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.~~ ~~*Op. Modes: Not Applicable*~~
~~**PD-AU2** UNPLANNED rise in plant radiation levels.~~ ~~*Op. Modes: Not Applicable*~~	~~**PD-AA2** UNPLANNED rise in plant radiation levels that impedes plant access required to maintain spent fuel integrity.~~ ~~*Op. Modes: Not Applica*~~ble
~~**PD-SU1** UNPLANNED spent fuel pool temperature rise.~~ ~~*Op. Modes: Not Applicable*~~	
~~**PD-HU1** Confirmed SECURITY CONDITION or threat.~~ ~~*Op. Modes: Not Applicable*~~	~~**PD-HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.~~ ~~*Op. Modes: Not Applicable*~~
~~**PD-HU2** Hazardous event affecting SAFETY SYSTEM equipment necessary for spent fuel cooling.~~ ~~*Op.*~~ Modes~~*: Not Applicable*~~	
~~**PD-HU3** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO)UE.~~ ~~*Op. Modes: Not Applicable*~~	~~**PD-HA3** Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.~~ ~~*Op. Modes: Not Applicable*~~

~~Table intended for use by EAL developers. Inclusion in licensee documents is not required. NEI 99-01 (Revision 6) November 2012 C-3~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** Release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.~~

~~**Operating Mode Applicability:** Not Applicable~~

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

~~**Notes:**~~

- ~~The Emergency Director should declare the Unusual Event promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.~~
- ~~If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 60 minutes.~~
- ~~If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.~~

~~(1) Reading on **ANY** effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.~~

~~(2) Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.~~

~~**Basis:**~~

~~This IC addresses a potential decrease in the level of safety of the plant as indicated by a low-level radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release). It includes any gaseous or liquid radiological release, monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.~~

~~Nuclear power plants incorporate design features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, and to control and monitor intentional releases. The occurrence of an extended, uncontrolled radioactive release to the environment is indicative of degradation in these features and/or controls.~~

~~Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.~~

~~Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.~~

~~Releases should not be prorated or averaged. For example, a release exceeding 4 times release limits for 30 minutes does not meet the EAL.~~

~~EAL #1 – This EAL addresses radioactivity releases that cause effluent radiation monitor readings to exceed 2 times the limit established by a radioactivity discharge permit. This EAL will typically be associated with planned batch releases from non-continuous release pathways (e.g., radwaste, waste gas).~~

~~EAL #2 – This EAL addresses uncontrolled gaseous or liquid releases that are detected by sample analyses or environmental surveys, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.).~~
~~Escalation of the emergency classification level would be via IC PD-AA1.~~

~~**Developer Notes:**~~

~~The "site-specific effluent release controlling document" is the Radiological Effluent Technical Specifications (RETS) or, for plants that have implemented Generic Letter 89-0111, the Offsite Dose Calculation Manual (ODCM). These documents implement regulations related to effluent controls (e.g., 10 CFR Part 20 and 10 CFR Part 50, Appendix I). As appropriate, the RETS or ODCM methodology should be used for establishing the monitor thresholds for this IC.~~

~~Listed monitors should include the effluent monitors described in the RETS or ODCM.~~

~~Developers may also consider including installed monitors associated with other potential effluent pathways that are not described in the RETS or ODCM1213. If included, EAL values for these monitors should be determined using the most applicable dose/release limits presented in the RETS or ODCM. It is recognized that a calculated EAL value may be below what the monitor can read; in that case, the monitor does not need to be included in the list. Also, some monitors may not be governed by Technical Specifications or other license-related related requirements; therefore, it is important that the associated EAL and basis section clearly identify any limitations on the use or availability of these monitors.~~

~~Some sites may find it advantageous to address gaseous and liquid releases with separate EALs.~~

~~Radiation monitor readings should reflect values that correspond to a radiological release exceeding 2 times a release control limit. The controlling document typically describes methodologies for determining effluent radiation monitor setpoints; these methodologies should be used to determine EAL values. In cases where a methodology is not adequately defined, developers should determine values consistent with effluent control regulations (e.g., 10 CFR Part 20 and 10 CFR Part 50 Appendix I) and related guidance.~~

~~11 *Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program*~~
~~12 This includes consideration of the effluent monitors described in the site emergency plan section(s) which address the requirements of 10 CFR 50.47(b)(8) and (9).~~
~~13 Developers should keep in mind the requirements of 10 CFR 50.54(q) and the guidance provided by INPO related to emergency response equipment when considering the addition of other effluent monitors.~~

~~For EAL #1 – Values in this EAL should be 2 times the setpoint established by the radioactivity discharge permit to warn of a release that is not in compliance with the specified limits. Indexing the value in this manner ensures consistency between the EAL and the setpoint established by a specific discharge permit.~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available.~~
~~For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.~~

~~Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.~~

~~ECL Assignment Attributes: 3.1.1.B NEI 99-01 (Revision 6) November 2012 C-6~~

~~**PD-AU2**~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** UNPLANNED rise in plant radiation levels.~~

~~**Operating Mode Applicability:** Not Applicable~~

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

~~(1) a. UNPLANNED water level drop in the spent fuel pool as indicated by ANY of the following:~~

~~(site-specific level indications).~~

~~**AND**~~

~~b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors.~~

~~(site-specific list of area radiation monitors).~~

~~(2) Area radiation monitor reading or survey result indicates an UNPLANNED rise of 25 mR/hr over NORMAL LEVELS.~~

~~**Basis:**~~

~~This IC addresses elevated plant radiation levels caused by a decrease in water level above irradiated (spent) fuel or other UNPLANNED events. The increased radiation levels are indicative of a minor loss in the ability to control radiation levels within the plant or radioactive materials. Either condition is a potential degradation in the level of safety of the plant.~~

~~A water level decrease will be primarily determined by indications from available level instrumentation. Other sources of level indications may include reports from plant personnel or video camera observations (if available). A significant drop in the water level may also cause an increase in the radiation levels of adjacent areas that can be detected by monitors in those locations.~~

~~The effects of planned evolutions should be considered. Note that EAL #1 is applicable only in cases where the elevated reading is due to an UNPLANNED water level drop. EAL #2 excludes radiation level increases that result from planned activities such as use of radiographic sources and movement of radioactive waste materials.~~

~~Escalation of the emergency classification level would be via IC PD-AA1 or PD-AA2.~~

~~**Developer Notes:**~~

~~For EAL #1 - Site-specific indications may include instrumentation values such as water level and area radiation monitor readings, and personnel reports. If available, video cameras may allow for remote observation. Depending on available instrumentation, the declaration may also be based on indications of water makeup rate and/or decreases in the level of a water storage tank.~~

~~For EAL #2 - The specified value of 25 mR/hr may be set to another value for a specific application with appropriate justification.~~

~~ECL Assignment Attributes: 3.1.1.B~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** UNPLANNED spent fuel pool temperature rise.~~

~~**Operating Mode Applicability:** Not Applicable~~

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

~~(1) UNPLANNED spent fuel pool temperature rise to greater than (site-specific ° F).~~

~~**Basis:**~~

~~This IC addresses a condition that is a precursor to a more serious event and represents a potential degradation in the level of safety of the plant. If uncorrected, boiling in the pool will occur, and result in a loss of pool level and increased radiation levels.~~

~~Escalation of the emergency classification level would be via IC PD-AA1 or PD-AA2.~~

~~**Developer Notes:**~~

~~The site-specific temperature should be chosen based on the starting point for fuel damage calculations in the SAR. Typically, this temperature is 125° to 150° F. Spent Fuel Pool temperature is normally maintained well below this point thus allowing time to correct the cooling system malfunction prior to classification.~~

~~ECL Assignment Attributes: 3.1.1.A NEI 99-01 (Revision 6) November 2012 C-9~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** Confirmed SECURITY CONDITION or threat.~~

~~**Operating Mode Applicability:** Not Applicable~~

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

~~(1) A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the (site-specific security shift supervision).~~

~~(2) Notification of a credible security threat directed at the site.~~

~~(3) A validated notification from the NRC providing information of an aircraft threat.~~

~~**Basis:**~~

~~This IC addresses events that pose a threat to plant personnel or the equipment necessary to maintain cooling of spent fuel, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under IC PD-HA1.~~

~~Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and OROs.~~

~~Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.~~

~~EAL #1 references (site-specific security shift supervision) because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.39 information.~~

~~EAL #2 addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with (site-specific procedure).~~

~~EAL #3 addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with (site-specific procedure).~~

~~Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a~~

~~potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.~~

~~Escalation of the emergency classification level would be via IC PD-HA1.~~

~~**Developer Notes:**~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

~~The (site-specific procedure) is the procedure(s) used by Control Room and/or Security personnel to determine if a security threat is credible, and to validate receipt of aircraft threat information.~~

~~Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.~~

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~ECL Assignment Attributes: 3.1.1.A NEI 99-01 (Revision 6) November 2012 C-11~~

~~**PD-HU2**~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** Hazardous event affecting SAFETY SYSTEM equipment necessary for spent fuel cooling.~~

~~**Operating Mode Applicability:** Not Applicable~~

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

~~(1) a. The occurrence of **ANY** of the following hazardous events:~~

- ~~Seismic event (earthquake)~~
- ~~Internal or external flooding event~~
- ~~High winds or tornado strike~~
- ~~FIRE~~
- ~~EXPLOSION~~
- ~~(site-specific hazards)~~
- ~~Other events with similar hazard characteristics as determined by the Shift Manager~~

~~**AND**~~

~~b. The event has damaged at least one train of a SAFETY SYSTEM needed for spent fuel cooling.~~

~~**AND**~~

~~c. The damaged SAFETY SYSTEM train(s) cannot, or potentially cannot, perform its design function based on **EITHER**:~~

- ~~Indications of degraded performance~~
- ~~VISIBLE DAMAGE~~

~~**Basis:**~~

~~This IC addresses a hazardous event that causes damage to at least one train of a SAFETY SYSTEM needed for spent fuel cooling. The damage must be of sufficient magnitude that the system(s) train cannot, or potentially cannot, perform its design function. This condition reduces the margin to a loss or potential loss of the fuel clad barrier, and therefore represents a potential degradation of the level of safety of the plant.~~

~~For EAL 1.c, the first bullet addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available.~~

~~For EAL 1.c, the second bullet addresses damage to a SAFETY SYSTEM train that is not in service/operation or readily apparent through indications alone. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.~~

~~Escalation of the emergency classification level could, depending upon the event, be based on any of the Alert ICs; PD-AA1, PD-AA2, PD-HA1 or PD-HA3.~~

~~Developer Notes:~~

~~For (site-specific hazards), developers should consider including other significant, site-specific hazards to the bulleted list contained in EAL 1.a (e.g., a seiche).~~

~~Nuclear power plant SAFETY SYSTEMS are comprised of two or more separate and redundant trains of equipment in accordance with site-specific design criteria.~~

~~ECL Assignment Attributes: 3.1.1.A and 3.1.1C~~

~~**ECL:** Notification of Unusual Event~~

~~**Initiating Condition:** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO)UE.~~

~~**Operating Mode Applicability:** Not Applicable~~

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

~~(1) Other conditions exist which in the judgment of the Emergency Director indicate that 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 offsite response or monitoring are expected unless further degradation of safety systems occurs~~.

~~**Basis:**~~

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

~~**ECL:** Alert~~

~~**Initiating Condition:** Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.~~

~~**Operating Mode Applicability:** Not Applicable~~

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

~~**Notes:**~~

- ~~The Emergency Director should declare the Alert promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.~~

~~If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.~~

~~If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.~~

~~The pre-calculated effluent monitor values presented in EAL #1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available~~.

~~(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer:~~

~~(site-specific monitor list and threshold values)~~

~~(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point).~~

~~(3) Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point) for one hour of exposure.~~

~~(4) Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point):~~

- ~~Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer.~~
- ~~Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.~~

~~**Basis:**~~

~~This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both~~

~~monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a~~

~~radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).~~

~~Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.~~

~~The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE. Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.~~

~~Developer Notes:~~

~~While this IC may not be met absent challenges to the cooling of spent fuel, it provides classification diversity and may be used to classify events that would not reach the same ECL based on plant conditions alone.~~

~~The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of these IC/EALs, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR § 20, is used in lieu of "…sum of EDE and CEDE….".~~

~~The EPA PAG guidance provides for the use adult thyroid dose conversion factors; however, some states have decided to base protective actions on child thyroid CDE. Nuclear power plant ICs/EALs need to be consistent with the protective action methodologies employed by the States within their EPZs. The thyroid CDE dose used in the IC and EALs should be adjusted as necessary to align with State protective action decision-making criteria.~~

~~The "site-specific monitor list and threshold values" should be determined with consideration of the following:~~

- ~~Selection of the appropriate installed gaseous and liquid effluent monitors.~~
- ~~The effluent monitor readings should correspond to a dose of 10 mrem TEDE or 50 mrem thyroid CDE at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.~~
- ~~Monitor readings will be calculated using a set of assumed meteorological data or atmospheric dispersion factors; the data or factors selected for use should be the same as those employed to calculate the monitor readings for IC PD-AU1.~~
- ~~The calculation of monitor readings will also require use of an assumed release isotopic mix; the selected mix should be the same as that employed to calculate monitor readings for IC PD-AU1.~~
- ~~Depending upon the methodology used to calculate the EAL values, there may be overlap of some values between different ICs. Developers will need to address this overlap by adjusting these values in a manner that ensures a logical escalation in the ECL.~~

~~The "site-specific dose receptor point" is the distance(s) and/or locations used by the licensee to distinguish between on-site and offsite doses. The selected distance(s) and/or locations should reflect the content of the emergency plan, and the procedural methodology used to determine offsite doses and Protective Action Recommendations. The variation in selected dose receptor points means there may be some differences in the distance from the release point to the calculated dose point from site to site.~~

~~Developers should research radiation monitor design documents or other information sources to ensure that 1) the EAL value being considered is within the usable response and display range of the instrument, and 2) there are no automatic features that may render the monitor reading invalid (e.g., an auto-purge feature triggered at a particular indication level).~~

~~It is recognized that the condition described by this IC may result in a radiological effluent value beyond the operating or display range of the installed effluent monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.~~

~~Although the IC references TEDE, field survey results are generally available only as a "whole body" dose rate. For this reason, the field survey EAL specifies a "closed window" survey reading.~~

~~Indications from a real-time dose projection system are not included in the generic EALs. Many licensees do not have this capability. For those that do, the capability may not be within the scope of the plant Technical Specifications. A licensee may request to include an EAL using real-time dose projection system results; approval will be considered on a case-by-case basis.~~

~~Indications from a perimeter monitoring system are not included in the generic EALs. Many licensees do not have this capability. For those that do, these monitors may not be controlled and maintained to the same level as plant equipment, or within the scope of the plant Technical Specifications. In addition, readings may be influenced by environmental or other factors. A licensee may request to include an EAL using a perimeter monitoring system; approval will be considered on a case-by-case basis.~~

~~ECL Assignment Attributes: 3.1.2.C~~

~~**ECL:** Alert~~

~~**Initiating Condition:** UNPLANNED rise in plant radiation levels that impedes plant access required to maintain spent fuel integrity.~~

~~**Operating Mode Applicability:** Not Applicable~~

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

~~(1) UNPLANNED dose rate greater than 15 mR/hr in **ANY** of the following areas requiring continuous occupancy to maintain control of radioactive material or operation of systems needed to maintain spent fuel integrity:~~

~~(site-specific area list)~~

~~(2) UNPLANNED Area Radiation Monitor readings or survey results indicate a rise by 100 mR/hr over NORMAL LEVELS that impedes access to **ANY** of the following areas needed to maintain control of radioactive material or operation of systems needed to maintain spent fuel integrity.~~

~~(site-specific area list~~)

~~**Basis:**~~

~~This IC addresses increased radiation levels that impede necessary access to areas containing equipment that must be operated manually or that requires local monitoring, in order to maintain systems needed to maintain spent fuel integrity. As used here, 'impede' includes hindering or interfering, provided that the interference or delay is sufficient to significantly threaten necessary plant access. It is this impaired access that results in the actual or potential substantial degradation of the level of safety of the plant.~~

~~This IC does not apply to anticipated temporary increases due to planned events.~~

~~**Developer Notes:**~~

~~The value of 15mR/hr is derived from the GDC 19 value of 5 rem in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737, *Clarification of TMI Action Plan Requirements*, provides that the 15 mR/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an Alert.~~

~~The specified value of 100 mR/hr may be set to another value for a specific application with appropriate justification.~~

~~ECL Assignment Attributes: 3.1.2.C~~

~~**ECL:** Alert~~

~~**Initiating Condition:** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.~~

~~**Operating Mode Applicability:** Not Applicable~~

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

~~(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the (site-specific security shift supervision).~~
~~(2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.~~

~~**Basis:**~~

~~This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.~~

~~Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.~~
~~Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.~~

~~As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of Offsite Response Organizations, allowing them to be better prepared should it be necessary to consider further actions.~~

~~This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.~~

~~EAL #1 is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes any action directed against an ISFSI that is located within the OWNER CONTROLLED AREA.~~

~~EAL #2 addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and OROs are in a heightened~~

~~state of readiness. This EAL is met when the threat-related information has been validated in accordance with (site-specific procedure).~~

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

~~In some cases, it may not be readily apparent if an aircraft impact within the OWNER CONTROLLED AREA was intentional (i.e., a HOSTILE ACTION). It is expected, although not certain, that notification by an appropriate Federal agency to the site would clarify this point. In this case, the appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. The emergency declaration, including one based on other ICs/EALs, should not be unduly delayed while awaiting notification by a Federal agency.~~

~~Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.~~

~~**Developer Notes:**~~

~~The (site-specific security shift supervision) is the title of the on-shift individual responsible for supervision of the on-shift security force.~~

~~Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.~~

~~With due consideration given to the above developer note, EALs may contain alpha or numbered references to selected events described in the Security Plan and associated implementing procedures. Such references should not contain a recognizable description of the event. For example, an EAL may be worded as "Security event #2, #5 or #9 is reported by the (site-specific security shift supervision)."~~

~~See the related Developer Note in Appendix B, Definitions, for guidance on the development of a scheme definition for the OWNER CONTROLLED AREA.~~

~~ECL Assignment Attributes: 3.1.2.D~~

~~**ECL:** Alert~~

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

~~**Operating Mode Applicability:** Not Applicable~~

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

~~(1) Other conditions exist which in the judgment of the Emergency Director indicate that 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 Protective Action Guideline exposure levels.~~

~~**Basis:**~~

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

ATTACHMENT 3
Deviation/Difference Justification Matrix

STPEGS Emergency Action Level Deviation, Difference and Justification Matrix Rev. 0

NEI 99-01 Rev. 6 Implementation

APRIL 2014

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

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GENERAL COMMENTS

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

Section	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
GLOBAL #1	References to NEI 99-01	Replaced with STPEGS	Difference	Convert generic guidance to STPEGS specific.
GLOBAL #2	Defined terms in Appendix B; Title Case	Defined terms in Appendix B; Upper Case	Difference	All defined terms in Appendix B used in the document are in upper case (CAPs) to indicate that the terms are defined.
GLOBAL #3	Notification of Unusual Event (NOUE)	UNUSUAL EVENT (UE)	Difference	Maintain continuity with previous practice at STPEGS
GLOBAL #4	BWR specific references	BWR references removed	Difference	STPEGS is a PWR
GLOBAL #5	Recognition Category A-Abnormal Radiation Levels/Radiological Effluent category and Emergency Action Levels; AU, AA, AS, and AG	Recognition Category R-Abnormal Radiation Levels/Radiological Effluent category and Emergency Action Levels; RU, RA, RS, and RG	Difference	STPEGS implemented the optional designation of "R" for radiological related items to maintain continuity with previous practice at STPEGS.
GLOBAL #6	Operating Mode Applicability lists mode names (i.e., Power Operation, Startup)	Operating Mode Applicability lists mode numbers (i.e., Modes 1 and 2)	Difference	Mode numbers used for consistency with STPEGS procedures and training.
GLOBAL #7	Developer's Notes	Developer's Notes deleted	Difference	Developer's notes are not reflected in the implementation of the EALs.
GLOBAL #8	Parameters or indications listed in EALs	Some parameters or indications listed in EALs were placed in tables.	Difference	Tables were created to consolidate data for use in STPEGS EAL Wall Boards.
GLOBAL #9	Site specific information or indication statements	Site specific information or indication statements were replaced with STPEGS information or indications where applicable and the statement deleted.	Difference	Compliance with intent of the guidance.
GLOBAL #10	EAL Basis	STPEGS EAL Selection Basis statements and References were added to supplement generic NEI Basis.	Difference	Supplemented generic Basis with STPEGS specific information and references within each EAL Recognition category.

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

Section	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
GLOBAL #11	Example EAL statement	"Example" deleted from statement	Difference	In adopting the EAL, the "example" status is no longer applicable.
GLOBAL #12	The following terms: "all, any" are sometimes capitalized and/or bolded in ICs and EALs	Consistently capitalized and bolded the following terms: "all, any" in ICs and EALs.	Difference	Capitalized and bolded conditional terms in ICs and EALs for consistency.
COVER PAGE	Development of Emergency Action Levels for Non-Passive Reactors	Technical Basis Document for the Development of STPEGS Emergency Action Levels	Difference	Indicate that the generic NEI guidance is now STPEGS specific
Introduction	Acknowledgments, Notice and Executive Summary	Deleted	Difference	Not Applicable to STPEGS
TOC	1. Regulatory Background	1.Development of Emergency Action Levels	Difference	Title change
TOC	1.1 Operating Reactors	1.1 Regulatory Background	Difference	Title change
TOC	1.2 Permanently Defueled Station	Deleted section	Difference	Not Applicable to STPEGS
TOC	1.3 Independent Spent Fuel Storage Installation (ISFSI)	1.2 Independent Spent Fuel Storage Installation (ISFSI)	Difference	Re-numbered
TOC	1.4 NRC Order EA-12-051	1.3 NRC Order EA-12-051	Difference	Re-numbered
TOC	1.5 Applicability of Advance and Small Modular Reactor Designs	Deleted section	Difference	Not Applicable to STPEGS
TOC	3.Design of the NEI 99-01 Emergency Classification Scheme	3. Design of the STPEGS Emergency Classification Scheme	Difference	Title Change
TOC	3.3 NSSS Design Differences	3.3 STPEGS Design Considerations	Difference	Title Change
TOC	4.0 Site-Specific Scheme Development	4.0 STPEGS Scheme Development	Difference	Title change
TOC	4.1 General Implementation Guidance	4.1 General Development Process	Difference	Title change
TOC	4.4; 4.5; 4.6; 4.8	Deleted sections	Difference	These sections contain general developer guidance for implementing document. Not relevant to STPEGS EAL development.
TOC	4.7 Developer and User Feedback	4.4 References to STPEGS AOPs and EOPs	Difference	Title change and re-numbered
TOC	Appendix C-Permanently Defueled Station ICs/EALs	Deleted	Difference	Not Applicable to STPEGS

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

Section	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
1.2	This IC is not applicable to installations or facilities that may process and/or repackage spent fuel (e.g., a Monitored Retrievable Storage Facility or an ISFSI at a spent fuel processing facility).	Statement deleted	Difference	The STPEGS ISFSI is not located at a processing facility nor is the facility a Monitored Retrieval Storage Facility.
1.4	Bottom paragraph referring to regulatory process for emergency plan changes	Deleted paragraph	Difference	Not applicable to EAL development.
3.1.2 (D)	"including those directed at an Independent Spent Fuel Storage Installation (ISFSI)"	Deleted text		STPEG's ISFSI is inside the PROTECTED AREA.
3.1.5.1		Added text: "STP is an Alternate AC plant and a Station Blackout battery copying analysis is not required. Nonetheless, a 125 VDC Battery Four Hour Coping Analysis was conducted and provides a basis for the time-based escalation path from a SITE AREA EMERGENCY to a GENERAL EMERGENCY."	Difference	STPEGS is an Alternate AC plant, and although not required, conducted a SBO battery coping analysis.
3.1.5	"the site specific coping period"	"four hours"	Difference	Inserted STPEGS coping period analysis.
3.3	Text referring to NSSS design differences for various types or nuclear plants; Developer guidance	Deleted and replaced with NSSS STPEGS design and informational considerations when developing STPEGS EALS.	Difference	Inserted STPEGS specific information to consider when developing EALs.
3.5	Mode of Applicability Matrix; Typical PWR Operating Modes	Deleted "Permanently Defueled" section of matrix; replaced Typical PWR Operating Modes with STPEGS Operating Modes	Difference	STPGEGS is not permanently defueled; Inserted STPEGS Operating Modes to comply with the intent of the document.
4.1	Text specific to EAL developers	Deleted text	Difference	Replaced with text specific to the development of STPEGS ICs and EALs

ABNORMAL RAD LEVELS / RADIOACTIVE EFFLUENT ICS/EALS

<table>
<tr><th>RC</th><th>NEI 99-01 Rev. 6</th><th>South Texas Project</th><th>Change</th><th>Justification</th></tr>
<tr><td rowspan="7">AU1</td><td>Recognition Category: AU1</td><td>Recognition Category: RU1</td><td>Difference</td><td>See Global Comment #5</td></tr>
<tr><td>Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.</td><td>Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 Times the ODCM limits for 60 Minutes or longer.</td><td>Difference</td><td>See Global Comment #9.</td></tr>
<tr><td>Operating Mode of Applicability: All</td><td>Operating Mode of Applicability: ALL</td><td>Difference</td><td>See Global Comment #12</td></tr>
<tr><td>(1) Reading on ANY effluent radiation monitor greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer: (site-specific monitor list and threshold values corresponding to 2 times the controlling document limits)</td><td>(1) Reading on ANY of the following radiation monitors greater than the values listed in Table R1 column "UE" for 60 minutes or longer:
<table>
<tr><th colspan="6">Table R1: Effluent Monitors</th></tr>
<tr><th>Release Point</th><th>Monitor</th><th>GE</th><th>SAE</th><th>ALERT</th><th>UE</th></tr>
<tr><td>Unit Vent</td><td>RT-8010B</td><td>1.50 E+08 μCi/sec</td><td>1.50 E+07 μCi/sec</td><td>1.50 E+06 μCi/sec</td><td>1.40 E+05 μCi/sec</td></tr>
<tr><td>Main Steam Lines</td><td>RT-8046 thru 8049</td><td>4.00 E+02 μCi/cm^3</td><td>4.00 E+01 μCi/cm^3</td><td>4.00 E+00 μCi/cm^3</td><td>5.00 E-02 μCi/cm^3</td></tr>
</table></td><td>Difference</td><td>See Global Comment #8 and #9</td></tr>
<tr><td>(2) Reading on ANY effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.</td><td>(2) Reading on gaseous effluent radiation monitor RT-8010B greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.</td><td>Difference</td><td>Altered wording so that threshold only applies to gaseous effluents. STPEGS liquid effluent monitor setpoint is normally orders of magnitude below the ODCM limit.</td></tr>
<tr><td>(3) Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.</td><td>(3) Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the ODCM limits for 60 minutes or longer.</td><td>Difference</td><td>See Global Comment #9.</td></tr>
<tr><td></td><td></td><td></td><td>The intent and meaning of the IC and EALs are not altered.</td></tr>
</table>

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
AU2	Recognition Category: AU2	Recognition Category: RU2	Difference	See Global Comment #5
	Initiating Condition: UNPLANNED loss of water level above irradiated fuel.	Initiating Condition: UNPLANNED loss of water level above irradiated fuel.	Verbatim	
	Operating Mode of Applicability: All	Operating Mode of Applicability: **ALL**	Difference	See Global Comment #12
	(1) a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by ANY of the following: (site-specific level indications).	(1) a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following: 1. Annunciator alarm on lampbox 22M02 Window F-5 "SFP WATER LVL HI/LO" **OR** 2. Visual Observation **OR** 3. Both of the following with spent fuel in the ICSA: a. Annunciator alarm on lampbox 22M02 Window F-6 "SFP Trouble" (Mode 5 or 6 only) **AND** b. Plant Computer point FCLC1420 "REFLNG CAV LVL IN CNTMT" (ICSA Water Level HI/LO) is in alarm (Mode 5 or 6 only)	Difference	See Global Comment #9 and #12
	AND b. UNPLANNED increase in area radiation levels as indicated by ANY of the following radiation monitors. (site-specific list of area radiation monitors)	**AND** b. UNPLANNED rise in area radiation levels on **ANY** of the following area radiation monitors: • RE-8055 (68' RCB) Mode 5 and 6 only • RE-8099 (68' RCB) Mode 5 and 6 only • RE-8090 (68' FHB)		See Global Comment #9 and #12 The intent and meaning of the EAL is not altered.

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

<table>
<tr><th>RC</th><th>NEI 99-01 Rev. 6</th><th>South Texas Project</th><th>Change</th><th>Justification</th></tr>
<tr><td rowspan="6">AA1</td><td>Recognition Category: AA1</td><td>Recognition Category: RA1</td><td>Difference</td><td>See Global Comment #5</td></tr>
<tr><td>Initiating condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.</td><td>Initiating condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem THYROID CDE.</td><td>Difference</td><td>See Global Comment #2</td></tr>
<tr><td>Operating Mode of Applicability: All</td><td>Operating Mode Applicability: ALL</td><td>Difference</td><td>See Global Comment #12</td></tr>
<tr><td>(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer:
(site-specific monitor list and threshold values)</td><td>(1) Reading on ANY of the following radiation monitors greater than the values listed in Table R1 column “ALERT” for 15 minutes or longer:
<table><tr><th colspan="6">Table R1: Effluent Monitors</th></tr><tr><th>Release Point</th><th>Monitor</th><th>GE</th><th>SAE</th><th>ALERT</th><th>UE</th></tr><tr><td>Unit Vent</td><td>RT-8010B</td><td>1.50 E+08 μCi/sec</td><td>1.50 E+07 μCi/sec</td><td>1.50 E+06 μCi/sec</td><td>1.40 E+05 μCi/sec</td></tr><tr><td>Main Steam Lines</td><td>RT-8046 thru 8049</td><td>4.00 E+02 μCi/cm³</td><td>4.00 E+01 μCi/cm³</td><td>4.00 E+00 μCi/cm³</td><td>5.00 E-02 μCi/cm³</td></tr></table></td><td>Differences</td><td>See Global Comment #8, #9 and #12</td></tr>
<tr><td>(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point).</td><td>(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem THYROID CDE at or beyond the SITE BOUNDARY.</td><td>Difference</td><td>See Global Comment #9 and #2</td></tr>
<tr><td>(3) Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point) for one hour of exposure.</td><td>(3) Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem THYROID CDE at or beyond the SITE BOUNDARY for one hour of exposure.</td><td>Difference</td><td>See Global Comment #9 and #2</td></tr>
</table>

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
AA1	(4) Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point): • Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.	(4) Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate THYROID CDE greater than 50 mrem for one hour of inhalation.	Difference	See Global Comment #9 and #2 The intent and meaning of the EALs are not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
AA2	Recognition Category: AA2	Recognition Category: RA2	Difference	See Global Comment #5
	Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel.	Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel.	Verbatim	
	Operating Mode of Applicability: All	Operating Mode of Applicability: **ALL**	Difference	See Global Comment #12
	(1) Uncovery of irradiated fuel in the REFUELING PATHWAY.	(1) Uncovery of irradiated fuel in the REFUELING PATHWAY.	Verbatim	
	(2) Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by ANY of the following radiation monitors: (site-specific listing of radiation monitors, and the associated readings, setpoints and/or alarms)	(2) a. Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by **ANY** of the following FHB radiation monitor readings: • ARM (68' FHB), RE-8090 greater than 1,500 mR/hr • FHB Exhaust, RT-8035 or RT-8036 greater than 1.00 E-1 $\mu Ci/cm^3$. *Continued on next page*	Difference	See Global Comment #9 and #12 The threshold value of 1500 mR/hr for area radiation monitor RE-8090 is within the calibrated range of the instrument. FHB Exhaust Monitors RT-8035 and RT-8036 were included because they are sensitive to noble gasses which are expected to be present if irradiated fuel is damaged. The calculated monitor response is 3.8 $\mu Ci/cm^3$ and the high range of the monitors is 0.3 $\mu Ci/cm^3$. The threshold value of 0.1 $\mu Ci/cm^3$ is approximately 6 orders of magnitude above background and indicative of damaged irradiated fuel. It was selected because it is readily assessable and within the calibrated range of the monitors.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
AA2		**OR** b. Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by **ANY** of the follow RCB radiation monitors readings (Mode 5 or 6 only). • ARMS (68' RCB), RE-8055 or RE-8099 greater than 850 mR/hr.	Difference	See Global Comment #9.
		NOTE *EAL-3 is not applicable until the enhanced SFP level instrumentation is available for use.*	Difference	Note added to indicate this EAL is not applicable until SFP level instrumentation is available for use.
	(3) Lowering of spent fuel pool level to (site-specific Level 2 value). [See Developer Notes	(3) Lowering of spent fuel pool level to 49'-10" or lower.	Difference	See Global Comment #9. The intent and meaning of the EALs are not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
AA3	Recognition Category: AA3	Recognition Category: RA3	Difference	See Global Comment #5
	Initiating Condition: Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.	Initiating Condition: Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.	Verbatim	
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
	(1) Dose rate greater than 15 mR/hr in ANY of the following areas: • Control Room • Central Alarm Station • (other site-specific areas/rooms)	(1) Dose rate greater than 15 mR/hr in **ANY** of the following areas: • Control Room ARM (RE-8066) • Central Alarm Station (CAS) by radiation survey	Difference	See Global Comment #12 Other site-specific areas/rooms removed as there were no additional applicable areas.
	(2) An UNPLANNED event results in radiation levels that prohibit or impede access to any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode applicability identified)	(2) An UNPLANNED event results in radiation levels that prohibit or impede access to **ANY** of the following plant rooms or areas: • Mechanical/Electrical Auxiliary Building (MEAB) • Turbine Generator Building (TGB) • Isolation Valve Cubicle (IVC) • Fuel Handling Building (FHB) • Reactor Containment Building (RCB)	Difference	See Global Comment #9 and #12 The intent and meaning of the EAL is not altered.

<table>
<tr><th>RC</th><th>NEI 99-01 Rev. 6</th><th>South Texas Project</th><th>Change</th><th>Justification</th></tr>
<tr><td rowspan="7">AS1</td><td>Recognition Category: AS1</td><td>Recognition Category: RS1</td><td>Difference</td><td>See Global Comment #5</td></tr>
<tr><td>Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.</td><td>Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem THYROID CDE.</td><td>Difference</td><td>See Global Comment #2</td></tr>
<tr><td>Operating Mode Applicability: All</td><td>Operating Mode Applicability: ALL</td><td>Difference</td><td>See Global Comment #12</td></tr>
<tr><td>(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer:
(site-specific monitor list and threshold values)</td><td>(1) Reading on ANY of the following radiation monitors greater than the values listed in Table R1 column "SAE" for 15 minutes or longer:
<table>
<tr><th colspan="6">Table R1: Effluent Monitors</th></tr>
<tr><th>Release Point</th><th>Monitor</th><th>GE</th><th>SAE</th><th>ALERT</th><th>UE</th></tr>
<tr><td>Unit Vent</td><td>RT-8010B</td><td>1.50 E+08 µCi/sec</td><td>1.50 E+07 µCi/sec</td><td>1.50 E+06 µCi/sec</td><td>1.40 E+05 µCi/sec</td></tr>
<tr><td>Main Steam Lines</td><td>RT-8046 thru 8049</td><td>4.00 E+02 µCi/cm³</td><td>4.00 E+01 µCi/cm³</td><td>4.00 E+00 µCi/cm³</td><td>5.00 E-02 µCi/cm³</td></tr>
</table></td><td>Differences</td><td>See Global Comment #12.

See Global Comment #8 and #9</td></tr>
<tr><td>(2) Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond (site-specific dose receptor point).</td><td>(2) Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem THYROID CDE at or beyond the SITE BOUNDARY.</td><td>Difference</td><td>See Global Comment #9 and #2</td></tr>
<tr><td>(3) Field survey results indicate EITHER of the following at or beyond (site-specific dose receptor point):
• Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer.
• Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.</td><td>(3) Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY:
• Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer.
• Analyses of field survey samples indicate THYROID CDE greater than 500 mrem for one hour of inhalation.</td><td>Difference</td><td>See Global Comment #9 and #2</td></tr>
<tr><td></td><td></td><td></td><td>The intent and meaning of the EALs are not altered.</td></tr>
</table>

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
AS2	Recognition Category: AS2	Recognition Category: RS2	Difference	See Global Comment #5
	Initiating Condition: Spent fuel pool level at (site-specific Level 3 description).	Initiating Condition: Spent fuel pool level at 40'-4" or lower.	Difference	See Global Comment #9
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
		NOTE *EAL-1 is not applicable until the enhanced SFP level instrumentation is available for use.*	Difference	Note added to indicate this EAL is not applicable until SFP level instrumentation is available for use.
	(1) Lowering of spent fuel pool level to (site-specific Level 3 value).	(1) Lowering of spent fuel pool level to 40'- 4" or lower.	Difference	See Global Comment #9 The intent and meaning of the EAL is not altered.

<table>
<tr><th>RC</th><th>NEI 99-01 Rev. 6</th><th>South Texas Project</th><th>Change</th><th>Justification</th></tr>
<tr><td rowspan="6">AG1</td><td>Recognition Category: AG1</td><td>Recognition Category: RG1</td><td>Difference</td><td>See Global Comment #5</td></tr>
<tr><td>Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.</td><td>Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem THYROID CDE.</td><td>Difference</td><td>See Global Comment #2</td></tr>
<tr><td>Operating Mode Applicability: All</td><td>Operating Mode Applicability: ALL</td><td>Difference</td><td>See Global Comment #12</td></tr>
<tr><td>(1) Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer:
(site-specific monitor list and threshold values)</td><td>(1) Reading on ANY of the following radiation monitors greater than the values listed in Table R1 column "GE" for 15 minutes or longer:
<table>
<tr><th colspan="6">Table R1: Effluent Monitors</th></tr>
<tr><th>Release Point</th><th>Monitor</th><th>GE</th><th>SAE</th><th>ALERT</th><th>UE</th></tr>
<tr><td>Unit Vent</td><td>RT-8010B</td><td>1.50 E+08 µCi/sec</td><td>1.50 E+07 µCi/sec</td><td>1.50 E+06 µCi/sec</td><td>1.40 E+05 µCi/sec</td></tr>
<tr><td>Main Steam Lines</td><td>RT-8046 thru 8049</td><td>4.00 E+02 µCi/cm^3</td><td>4.00 E+01 µCi/cm^3</td><td>4.00 E+00 µCi/cm^3</td><td>5.00 E-02 µCi/cm^3</td></tr>
</table></td><td>Differences</td><td>See Global Comment #8 and #9</td></tr>
<tr><td>(2) Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond (site-specific dose receptor point).</td><td>(2) Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem THYROID CDE at or beyond the SITE BOUNDARY.</td><td>Difference</td><td>See Global Comment #9 and #2</td></tr>
<tr><td>(3) Field survey results indicate EITHER of the following at or beyond (site-specific dose receptor point):
• Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer.
• Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.</td><td>(3) Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY:
• Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer.
• Analyses of field survey samples indicate THYROID CDE greater than 5,000 mrem for one hour of inhalation.</td><td>Difference</td><td>See Global Comment #9 and #2

The intent and meaning of the EALs are not changed.</td></tr>
</table>

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
AG2	Recognition Category: AG2	Recognition Category: RG2	Difference	See Global Comment #5
	Initiating Condition: Spent fuel pool level cannot be restored to at least (site-specific Level 3 description) for 60 minutes or longer.	Initiating Condition: Spent fuel pool level cannot be restored to at least 40'- 4" for 60 minutes or longer.	Difference	See Global Comment #9
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
		NOTE *EAL-1 is not applicable until the enhanced SFP level instrumentation is available for use.*	Difference	Note added to indicate this EAL is not applicable until SFP level instrumentation is available for use.
	(1) Spent fuel pool level cannot be restored to at least (site-specific Level 3 value) for 60 minutes or longer.	(1) Spent fuel pool level cannot be restored to at least 40'- 4" for 60 minutes or longer.	Difference	See Global Comment #9 The intent and meaning of the EAL is not altered.

COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
CU1	Recognition Category: CU1	Recognition Category: CU1	Verbatim	
	Initiating Condition: UNPLANNED loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory for 15 minutes or longer.	Initiating Condition: Unplanned loss of RCS inventory for 15 minutes or longer.	Difference	See Global Comment #4
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 5, 6	Difference	See Global Comment #6
	(1) UNPLANNED loss of reactor coolant results in (reactor vessel/RCS [PWR] or RPV [BWR]) level less than a required lower limit for 15 minutes or longer.	(1) UNPLANNED loss of reactor coolant results in RCS level below the Reactor Vessel Flange for 15 minutes or longer.	Difference	See Global Comment #4
	(2) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored. AND	(2) a. RCS level cannot be monitored. **AND**	Difference	See Global Comment #4
	b. UNPLANNED increase in (site-specific sump and/or tank) levels.	b. UNPLANNED rise in **ANY** of the following sump or tank levels in Table C2:	Difference	See Global Comment #12
		Table C2: RCS Leakage • Containment Normal Sump • Pressurizer Relief Tank (PRT) • Reactor Coolant Drain Tank (RCDT) • MAB Sumps 1 thru 4 • Containment Penetration Area Sump • SIS/CSS Pump Compartment Sump	Difference	See Global Comment #8 and #9 The intent and meaning of the IC and EALs are not altered.

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
CU2	Recognition Category: CU2	Recognition Category: CU2	Verbatim	
	Initiating Condition: Loss of all but one AC power source to emergency buses for 15 minutes or longer.	Initiating Condition: Loss of **ALL** but one AC power source to emergency buses for 15 minutes or longer.	Difference	See Global Comment #12
	Operating Mode Applicability: Cold Shutdown, Refueling, Defueled	Operating Mode Applicability: 5, 6, Defueled	Difference	See Global Comment #6
	(1) a. AC power capability to (site-specific emergency buses) is reduced to a single power source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS.	(1) a. AC power capability to **ALL** three 4160V AC ESF Buses is reduced to a single power source for 15 minutes or longer. **AND** b. **ANY** additional single power source failure will result in loss of ALL AC power to SAFETY SYSTEMS.	Difference	See Global Comment #9 and #12 The intent and meaning of the EAL is not altered.

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
CU3	Recognition Category: CU3	Recognition Category: CU3	Verbatim	
	Initiating Condition: UNPLANNED increase in RCS temperature.	Initiating Condition: UNPLANNED rise in RCS temperature.	Verbatim	
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 5, 6	Difference	See Global Comment #6
	(1) UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit).	(1) UNPLANNED rise in RCS temperature to greater than 200 °F (Tavg).	Difference	See Global Comment #9
	(2) Loss of ALL RCS temperature and (reactor vessel/RCS [PWR] or RPV [BWR]) level indication for 15 minutes or longer.	(2) Loss of **ALL** RCS temperature and RCS level indication for 15 minutes or longer.	Difference	See Global Comment #4 and #12 The intent and meaning of the EALs are not altered.

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
CU4	Recognition Category: CU4	Recognition Category: CU4	Verbatim	
	Initiating Condition: Loss of Vital DC power for 15 minutes or longer.	Initiating Condition: Loss of Vital DC power for 15 minutes or longer.	Verbatim	
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 5, 6	Difference	See Global Comment #6
	(1) Indicated voltage is less than (site-specific bus voltage value) on required Vital DC buses for 15 minutes or longer.	(1) Indicated voltage is less than 105.5 VDC on required Class 1E 125 VDC battery buses for 15 minutes or longer.	Difference	See Global Comment #9 and #12 The intent and meaning of the EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
CU5	Recognition Category: CU5	Recognition Category: CU5	Verbatim	
	Initiating Condition: Loss of all onsite or offsite communications capabilities.	Initiating Condition: Loss of **ALL** onsite or offsite communications capabilities.	Difference	See Global Comment #12
	Operating Mode Applicability: Cold Shutdown, Refueling, Defueled	Operating Mode Applicability: 5, 6, Defueled	Difference	See Global Comment #6
	(1) Loss of ALL of the following onsite communication methods: (site-specific list of communications methods)	(1) Loss of **ALL** of the following onsite communication methods in Table C4.	Difference	See Global Comment #9 and #12
	(2) Loss of ALL of the following ORO communications methods: (site-specific list of communications methods)	(2) Loss of **ALL** of the following Offsite Response Organization (ORO) communications methods in Table C4.	Difference	See Global Comment #9 and #12
	(3) Loss of ALL of the following NRC communications methods: (site-specific list of communications methods)	(3) Loss of **ALL** of the following NRC communications methods in Table C4.	Difference	See Global Comment #9 and #12
		Table C4 (below)	Difference	See Global Comment #8 The intent and meaning of the EALs are not altered.

Table C4: Communications Methods

METHOD	EAL-1 ONSITE	EAL-2 ORO	EAL-3 NRC
• Plant PA system	X		
• Plant Radios	X		
• Plant telephone system	X	X	X
• Satellite phones		X	X
• Direct line from Control Rooms to Bay City		X	X
• Microwave Lines to Houston		X	X
• Security radio to Matagorda County		X	
• Dedicated Ring-down lines		X	
• ENS line			X

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
CA1	Recognition Category: CA1	Recognition Category: CA1	Verbatim	
	Initiating Condition: Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory.	Initiating Condition: Loss of RCS inventory.	Difference	See Global Comment #4
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 5, 6	Difference	See Global Comment #6
	(1) Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory as indicated by level less than (site-specific level).	(1) Loss of RCS inventory as indicated by level less than 32 ft. 9 inch (+6 inches above hot leg centerline).	Differences	See Global Comment #4 and #9
	(2) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored for 15 minutes or longer AND	(2) a. RCS level cannot be monitored for 15 minutes or longer **AND**	Difference	See Global Comment #4 and #9
	b. UNPLANNED increase in (site-specific sump and/or tank) levels due to a loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory.	b. UNPLANNED rise in **ANY** of the following sump or tank levels in Table C2 due to a loss of reactor vessel/ RCS inventory:	Differences	See Global Comment #4 and #9
		Table C2: RCS Leakage • Containment Normal Sump • Pressurizer Relief Tank (PRT) • Reactor Coolant Drain Tank (RCDT) • MAB Sumps 1 thru 4 • Containment Penetration Area Sump • SIS/CSS Pump Compartment Sump	Difference	See Global Comment #8 The intent and meaning of the EALs are not altered.

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
CA2	Recognition Category: CA2	Recognition Category: CA2	Verbatim	
	Initiating Condition: Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.	Initiating Condition: Loss of **ALL** offsite and **ALL** onsite AC power to emergency buses for 15 minutes or longer.	Difference	See Global Comment #12
	Operating Mode Applicability: Cold Shutdown, Refueling, Defueled	Operating Mode Applicability: 5, 6, Defueled	Difference	See Global Comment #6
	(1) Loss of ALL offsite and ALL onsite AC Power to (site-specific emergency buses) for 15 minutes or longer.	(1) Loss of **ALL** offsite and **ALL** onsite AC Power to **ALL** three 4160V AC ESF Busses for 15 minutes or longer.	Difference	See Global Comment #9 and #12 The intent and meaning of the EAL is not altered.

<table>
<tr><th>RC</th><th>NEI 99-01 Rev. 6</th><th>South Texas Project</th><th>Change</th><th>Justification</th></tr>
<tr><td rowspan="6">CA3</td><td>Recognition Category: CA3</td><td>Recognition Category: CA3</td><td>Verbatim</td><td></td></tr>
<tr><td>Initiating Condition: Inability to maintain the plant in cold shutdown.</td><td>Initiating Condition: Inability to maintain the plant in cold shutdown.</td><td>Verbatim</td><td></td></tr>
<tr><td>Operating Mode Applicability: Cold Shutdown, Refueling</td><td>Operating Mode Applicability: 5, 6</td><td>Difference</td><td>See Global Comment #6</td></tr>
<tr><td>(1) UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit) for greater than the duration specified in the following table.</td><td>(1) UNPLANNED rise in RCS temperature greater than 200 °F (Tavg) for greater than the duration specified in Table C3:</td><td>Difference</td><td>See Global Comment #9</td></tr>
<tr><td>
<table>
<tr><th colspan="3">Table: RCS Heat-up Duration Thresholds</th></tr>
<tr><th>RCS Status</th><th>Containment Closure Status</th><th>Heat-up Duration</th></tr>
<tr><td>Intact (but not at reduced inventory [PWR])</td><td>Not applicable</td><td>60 minutes*</td></tr>
<tr><td rowspan="2">Not intact (or at reduced inventory [PWR])</td><td>Established</td><td>20 minutes*</td></tr>
<tr><td>Not Established</td><td>0 minutes</td></tr>
<tr><td colspan="3">* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.</td></tr>
</table>
</td><td>
<table>
<tr><th colspan="3">Table C3: RCS Heat-up Duration Thresholds</th></tr>
<tr><th>RCS Status</th><th>Containment Closure Status</th><th>Heat-up Duration</th></tr>
<tr><td>Intact (but not at reduced inventory)</td><td>Not applicable</td><td>60 minutes*</td></tr>
<tr><td rowspan="2">Not intact (or at reduced inventory)</td><td>Established</td><td>20 minutes*</td></tr>
<tr><td>Not Established</td><td>0 minutes</td></tr>
<tr><td colspan="3">* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.</td></tr>
</table>
</td><td>Differences</td><td>See Global Comment #4 and #8
Table labeled "Table C3."</td></tr>
<tr><td>(2) UNPLANNED RCS pressure increase greater than (site-specific pressure reading). (This EAL does not apply during water-solid plant conditions. [PWR])</td><td>(2) UNPLANNED RCS pressure rise greater than 10 psig. (This EAL does not apply during water-solid plant conditions.)</td><td></td><td>See Global Comment #4 and #9

The intent and meaning of the EALs are not altered.</td></tr>
</table>

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
CA6	Recognition Category: CA6	Recognition Category: CA6	Verbatim	
	Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Verbatim	
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 5, 6	Difference	See Global Comment #6
	(1) a. The occurrence of **ANY** of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • (site specific hazards) • Other events with similar hazard characteristics as determined by the Shift Manager	(1) a. The occurrence of **ANY** of the following hazardous events in Table C5: Table C5: Hazardous Events • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Predicted or actual breach of Main Cooling Reservoir retaining dike along the North Wall. • Other events with similar hazard characteristics as determined by the Shift Manager	Differences	See Global Comment #8 Table labeled Table C5 See Global Comment #9
	AND b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode.	**AND** b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode.	Verbatim	
	OR 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	**OR** 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	Verbatim	The intent and meaning of the EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
CS1	Recognition Category: CS1	Recognition Category: CS1	Verbatim	
	Initiating Condition: Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory affecting core decay heat removal capability.	Initiating Condition: Loss of RCS inventory affecting core decay heat removal capability.	Difference	See Global Comment #4
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 5, 6	Difference	See Global Comment #6
	(1) a. CONTAINMENT CLOSURE not established. **AND** b. (Reactor vessel/RCS [PWR] or RPV [BWR]) level less than (site-specific level).	(1) a. CONTAINMENT CLOSURE not established. **AND** b. RCS level less than 33% of plenum.	Differences	See Global Comment #4 and #9
	(2) a. CONTAINMENT CLOSURE established. **AND** b. (Reactor vessel/RCS [PWR] or RPV [BWR]) level less than (site-specific level).		Difference	EAL-2 from NEI 99-01 Rev.6 was removed, per developer notes; due to inability to monitor level as low as the threshold required for the reactor vessel/RCS level.
	(3) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored for 30 minutes or longer. **AND**	(2) a. RCS level cannot be monitored for 30 minutes or longer. **AND**	Difference	See Global Comment #4 EAL-3 re-numbered to EAL-2.
	b. Core uncovery is indicated by **ANY** of the following: (Site-specific radiation monitor) reading greater than (site-specific value)	b. Core uncovery as indicated by **ANY** of the following: • Reactor Containment Building 68'-0" Area Radiation Monitor RE-8055 or RE-8099 reading greater than 9,000 mR/hr *Continued on next page*	Difference	The calculated monitor response to core uncovery exceeds the upper range of the radiation monitors (10 R/hr). The threshold value of 9,000 mR/hr with the reactor head on is within the upper end of the calibrated range of the detector and corresponds to an RCS level approximately 8 inches above the top of the active fuel. These monitor readings in conjunction with the other indications of this EAL allow for accurate assessment of the EAL.

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
CS1	• Erratic source range monitor indication [PWR]	• Erratic source range monitor indication	Difference	See Global Comment #4
	• UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery • (Other site-specific indications)	• UNPLANNED rise in **ANY** of the sump or tank levels in Table C2 of sufficient magnitude to indicate core uncovery: Table C2: RCS Leakage • Containment Normal Sump • Pressurizer Relief Tank (PRT) • Reactor Coolant Drain Tank (RCDT) • MAB Sumps 1 thru 4 • Containment Penetration Area Sump • SIS/CSS Pump Compartment Sump	Difference	See Global Comment #8 and #9 The intent and meaning of the EALs are not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
CG1	Recognition Category: CG1	Recognition Category: CG1	Verbatim	
	Initiating Condition: Loss of (reactor vessel/RCS [PWR] or RPV [BWR]) inventory affecting fuel clad integrity with containment challenged.	Initiating Condition: Loss of RCS inventory affecting fuel clad integrity with containment challenged.	Difference	See Global Comment #4
	Operating Mode Applicability: Cold Shutdown, Refueling	Operating Mode Applicability: 5, 6	Difference	See Global Comment #6
	(1) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level less than (site-specific level) for 30 minutes or longer. **AND** b. ANY indication from the Containment Challenge Table (see below).		Difference	EAL 1 was removed; per developer notes, due to the inability to monitor reactor vessel/RCS level as low as the threshold required.
	(2) a. (Reactor vessel/RCS [PWR] or RPV [BWR]) level cannot be monitored for 30 minutes or longer. AND	(1) a. RCS level cannot be monitored for 30 minutes or longer. **AND**	Differences	See Global Comment #4 EAL 2 re-numbered to EAL 1.
	b. Core uncovery is indicated by ANY of the following: • (Site-specific radiation monitor) reading greater than (site-specific value)	b. Core uncovery is indicated by **ANY** of the following: • Reactor Containment Building 68'-0" Area Radiation Monitor RE-8055 or RE-8099 reading greater than 9,000 mR/hr	Difference	The calculated monitor response to core uncovery exceeds the upper range of the radiation monitors (10 R/hr). The threshold value of 9,000 mR/hr is within the upper end of the calibrated range of the detector and corresponds to an RCS level approximately 8 inches above the top of the active fuel. These monitor readings in conjunction with the other indications of this EAL allow for accurate assessment of the EAL.
	• Erratic source range monitor indication [PWR]	• Erratic source range monitor indication	Difference	See Global Comment #4

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
CG1	• UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery	• UNPLANNED rise in **ANY** of the following sump or tank levels in Table C2 of sufficient magnitude to indicate core uncovery: **Table C2: RCS Leakage** • Containment Normal Sump • Pressurizer Relief Tank (PRT) • Reactor Coolant Drain Tank (RCDT) • MAB Sumps 1 thru 4 • Containment Penetration Area Sump • SIS/CSS Pump Compartment Sump	Differences	See Global Comment #8 and #9
	AND c. **ANY** indication from the Containment Challenge Table (see below).	**AND** c. **ANY** indication from the Table C1.	Difference	Syntax altered to refer to Table C1.
	Containment Challenge Table • CONTAINMENT CLOSURE not established* • (Explosive mixture) exists inside containment • UNPLANNED increase in containment pressure • Secondary containment radiation monitor reading above (site specific value) [BWR]	**Table C1: Containment Challenge** • CONTAINMENT CLOSURE not established within 30 minutes OR • ≥ 4% hydrogen exists inside containment OR • UNPLANNED rise in containment pressure	Differences	See Global Comment #4, #8 and #9
	* If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.		Verbatim	Intent and meaning of the IC and EALs are not altered.

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) ICS/EALS

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
E-HU1	Recognition Category: E-HU1	Recognition Category: E-HU1	Verbatim	
	Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY.	Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY.	Verbatim	
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
	(1) Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than (2 times the site-specific cask specific technical specification allowable radiation level) on the surface of the spent fuel cask.	(1) Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than: a. 60 mrem/hr (gamma + neutron) on the top surface of the spent fuel cask **OR** b. 600 mrem/hr (gamma + neutron) on the side surface of the spent fuel cask **OR** c. 7000 mrem/hr (gamma + neutron) on the side surface of the transfer cask.	Difference	See Global Comment #9 Intent and meaning of the EAL is not altered.

FISSION PRODUCT BARRIER ICS/EALS

The following section is configured in a manner that is different from the Fission Product Barrier Tables in the STPEGS EAL Technical Bases Document. Where the Technical Bases Document evaluates all three fission product barriers simultaneously for a specific sub-category, this matrix evaluates each fission product barrier individually for all sub-categories. The significance of this fact is that where the fission product barrier table in the Technical Bases Document moves vertically through the sub-categories, this matrix moves horizontally.

Fission Product Barrier Emergency Classifications							
NEI 99-01 Rev. 6			South Texas Project			Change	Justification
Table 9-F-3: PWR EAL Fission Product Barrier Table			Table 9-F-2: EAL Fission Product Barrier Table			Difference	Renumbered and re-labeled due to deletion of Table 9-F-2, BWR EAL Fission Product Barrier Table and eliminate redundancy in heading.
Figure 9-F-4: PWR EAL Fission Product Barrier Table			Figure 9-F-3: EAL Fission Product Barrier Table			Difference	Renumbered and re-labeled due to deletion of Table 9-F-2, BWR EAL Fission Product Barrier Table and eliminate redundancy in heading.
Basis Information For PWR EAL Fission Product Barrier Table 9-F-3 Developer Notes; pages 101-102			STPEGS is part of the Westinghouse Owners Group (WOG) and has adopted the WOG Emergency Response Guidelines (ERG). These guidelines employ the use of Critical Safety Function Status Trees (CSFST). Since STP has implemented the WOG ERGs, the guidance in NEI 99-01 allows the use of certain CSFST assessment results as EALs and fission product barrier loss/potential loss thresholds. This approach allows consistency between EOPs and emergency classifications.			Difference	STPEGS has elected to use the WOG CSFST where appropriate.
Basis Information For PWR EAL Fission Product Barrier Table 9-F-3 Developer Notes; pages 103-121			Changes indicate STPEGS specific values for the parameters identified in this section			Difference	Transform generic NEI 99-01 guidance into STPEGS specific application.
Alert	**Site Area Emergency**	**General Emergency**	**Alert**	**Site Area Emergency**	**General Emergency**	Verbatim	
Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier.	Loss or Potential Loss of any two barriers.	Loss of any two barriers and Loss or Potential Loss of the third barrier.	Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier.	Loss or Potential Loss of any two barriers.	Loss of any two barriers and Loss or Potential Loss of the third barrier.	Verbatim	

Thresholds for LOSS or POTENTIAL LOSS of Fuel Clad Barrier						
Table 9-F-2	**NEI 99-01 Rev. 6**		**South Texas Project**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
1. RCS or SG Tube Leakage	Not Applicable	A. RCS/reactor vessel level less than (site-specific level).	Not Applicable	A. Core Cooling - Orange entry conditions met	Difference	Loss A.- Threshold altered, per developer notes, to conform to STPEGS use of Westinghouse ERG CSFST terminology.
2. Inadequate Heat Removal	A. Core exit thermocouple readings greater than (site-specific temperature value).	A. Core exit thermocouple readings greater than (site-specific temperature value). **OR** B. Inadequate RCS heat removal capability via steam generators as indicated by (site-specific indications).	A. Core Cooling - Red entry conditions met	A. Core Cooling - Orange entry conditions met **OR** B. Heat Sink - Red entry conditions met	Difference Difference	Loss A.- Threshold altered, per developer notes, to conform to STPEGS use of Westinghouse ERG CSFST terminology. Potential Loss A. and B.- Thresholds altered, per developer notes, to conform to STP use of Westinghouse ERG CSFST terminology.
3. RCS Activity / Containment Radiation	A. Containment radiation monitor reading greater than (site-specific value). **OR** B. (Site-specific indications that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131).	Not Applicable	A 1. RCB Rad Monitor RT-8050 or RT-8051 greater than 140 R/hr **OR** 2. HATCH MONITOR greater than 300 mR/hr **OR** B. Sample analysis indicates that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131.	Not Applicable	Difference Difference	Loss A.- See Global Comment #9 Loss B.- See Global Comment #9 .

Thresholds for LOSS or POTENTIAL LOSS of Fuel Clad Barrier						
Table 9-F-2	**NEI 99-01 Rev. 6**		**South Texas Project**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
4. Containment Integrity or Bypass	Not Applicable	Not Applicable	Not Applicable	Not Applicable	Verbatim	
5. Other Indications	A. (site-specific as applicable)	A. (site-specific as applicable)	Not Applicable	Not Applicable	Difference	No additional site-specific indications are applicable.
6. Emergency Director Judgment	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	Verbatim	

Thresholds for LOSS or POTENTIAL LOSS of RCS Barrier						
Table 9-F-2	**NEI 99-01 Rev. 6**		**South Texas Project**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
1. RCS or SG Tube Leakage	A. An automatic or manual ECCS (SI) actuation is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube RUPTURE.	A. Operation of a standby charging (makeup) pump is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube leakage. **OR** B. RCS cooldown rate greater than (site-specific pressurized thermal shock criteria/limits defined by site-specific indications).	A. An automatic or manual ECCS (SI) actuation is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube RUPTURE.	A. Operation of a standby charging pump is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube leakage. **OR** B. Integrity – Red entry conditions met.	Difference	Potential Loss B.- Threshold altered, per developer notes, to conform to STPEGS use of Westinghouse ERG CSFST terminology.
2. Inadequate heat Removal	Not Applicable	A. Inadequate RCS heat removal capability via steam generators as indicated by (site-specific indications).	Not Applicable	A. Heat Sink - Red entry conditions met.	Difference	Potential Loss A - Threshold altered, per developer notes, to conform to STPEGS use of Westinghouse ERG CSFST terminology.
3. RCS Activity / Containment Radiation	A. Containment radiation monitor reading greater than (site-specific value).	Not Applicable	A. RCB Rad Monitor RT-8050 or RT-8051 reading greater than 2 R/hr.	Not Applicable	Difference	Threshold from combination of Calculation STPNOC013-CALC-004 result of 450 mR/hr and background reading of 1.5 R/hr.
4. Containment Integrity or Bypass	Not Applicable	Not Applicable	Not Applicable	Not Applicable	Verbatim	

STP EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

Thresholds for LOSS or POTENTIAL LOSS of RCS Barrier						
Table 9-F-2	**NEI 99-01 Rev. 6**		**South Texas Project**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
5. Other Indications	A. (site-specific as applicable)	A. (site-specific as applicable)	Not Applicable	Not Applicable	Difference	No additional site-specific indications are applicable.
6. Emergency Director Judgment	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	Verbatim	

STP EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

Thresholds for LOSS or POTENTIAL LOSS of Containment Barrier						
Table 9-F-2	**NEI 99-01 Rev. 6**		**South Texas Project**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
1. RCS or SG Tube Leakage	A. A leaking or RUPTURED SG is FAULTED outside of containment.	Not Applicable	A. A Leaking or RUPTURED SG is FAULTED outside of containment.	Not Applicable	Verbatim	
2. Inadequate heat Removal	Not Applicable	A 1 (Site-specific criteria for entry into core cooling restoration procedure) **AND** 2 Restoration procedure not effective within 15 minutes.	Not Applicable	A. Core Cooling – Red entry conditions met for 15 minutes or longer	Difference	Potential Loss A.- Threshold altered, per developer notes, to conform to STPEGS use of Westinghouse ERG CSFST terminology.
3. RCS Activity / Containment Radiation	Not Applicable	A. Containment radiation monitor reading greater than (site-specific value).	Not Applicable	A 1.RCB Rad Monitor RT-8050 or RT-8051 greater than 550 R/hr **OR** 2. HATCH MONITOR greater than 1200 mR/hr	Difference	See Global Comment #9
4. Containment Integrity or Bypass	A. Containment isolation is required **AND EITHER** of the following: 1. Containment integrity has been lost based on Emergency Director judgment.	A. Containment pressure greater than (site-specific value) **OR**	A. Containment isolation is required **AND EITHER** of the following: 1. Containment integrity has been lost based on Emergency Director judgment.	A. Containment - Red entry conditions met **OR**	Difference	Potential Loss A - Threshold per STPEGS use of Westinghouse ERG CSFST terminology.

Thresholds for LOSS or POTENTIAL LOSS of Containment Barrier						
Table 9-F-2	**NEI 99-01 Rev. 6**		**South Texas Project**		**Change**	**Justification**
Sub-Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**		
	OR 2. UNISOLABLE pathway from the containment to the environment exists. **OR**		**OR** 2. UNISOLABLE pathway from the containment to the environment exists. **OR**			
	B. Indications of RCS leakage outside of containment	B. Explosive mixture exists inside containment **OR**	B. Indications of RCS leakage outside of containment.	B. Explosive mixture exists inside containment ($H2 > 4\%$) **OR**	Difference	Potential Loss B. - See Global Comment #9
		C 1. Containment pressure greater than (site-specific pressure setpoint) **AND**		C 1. Containment pressure greater than 9.5 psig **AND**	Difference	Potential Loss C.1.- See Global Comment #9
		2. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer.		2. Less than one full train of Containment Spray is operating per design for 15 minutes or longer	Difference	Potential Loss C.2.- See Global Comment #9
5. Other Indications	A. (site-specific as applicable)	A. (site-specific as applicable)	Not Applicable	Not Applicable	Difference	No additional site-specific indications are applicable.
6. Emergency Director Judgment	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.	Verbatim	

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY ICS/EALS

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
HU1	Recognition Category: HU1	Recognition Category: HU1	Verbatim	
	Initiating Condition: Confirmed SECURITY CONDITION or threat.	Initiating Condition: Confirmed SECURITY CONDITION or threat.	Verbatim	
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
	(1) A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the (site-specific security shift supervision).	(1) A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by **ANY** of the following personnel in Table H1: Table H1: Security Supervision • Security Force Supervisor • Acting Security Manager • Security Manager	Difference	See Global Comment #9
	(2) Notification of a credible security threat directed at the site.	(2) Notification of a CREDIBLE SECURITY THREAT directed at the site.	Difference	See Global Comment #2
	(3) A validated notification from the NRC providing information of an aircraft threat.	(3) A validated notification from the NRC providing information of an aircraft threat.	Verbatim	
				Intent and meaning of the EALs are not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
HU2	Recognition Category: HU2	Recognition Category: HU2	Verbatim	
	Initiating Condition: Seismic event greater than OBE levels.	Initiating Condition: Seismic event greater than OBE levels.	Verbatim	
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
	(1) Seismic event greater than Operating Basis Earthquake (OBE) as indicated by: (site-specific indication that a seismic event met or exceeded OBE limits)	(1) a. **EITHER** of the following conditions exist: 1. "SEISMIC EVENT" alarm in Unit 1 Control Room (Lampbox 9M01, Window E-8). **OR** 2. Control Room personnel feel an actual or potential seismic event **AND** b. The occurrence of a seismic event is confirmed in a manner deemed appropriate by the Shift Manager or Emergency Director.	Difference	See Global Comment #9 The STPEGS Seismic Event Alarm setpoint is 0.02g in the vertical or horizontal position and the station design basis value for an OBE is 0.05g. Since the Seismic Event alarm is set at less than half of the OBE value, it cannot be used as the sole threshold value for determining whether or not STPEGS has experienced an OBE. STPEGS has implemented the alternative EAL described in NEI 99-01 Developer Notes in conjunction with using the installed indication. Intent and meaning of the EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
HU3	Recognition Category: HU3	Recognition Category: HU3	Verbatim	
	Initiating Condition: Hazardous event.	Initiating Condition: Hazardous event	Verbatim	
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
	(1) A tornado strike within the PROTECTED AREA.	(1) A tornado strike within the PROTECTED AREA.	Verbatim	
	(2) Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.	(2) Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.	Verbatim	
	(3) Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).	(3) Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).	Verbatim	
	(4) A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.	(4) A hazardous event that results in onsite conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.	Verbatim	
	(5) (Site-specific list of natural or technological hazard events)	(5) Predicted or actual breach of Main Cooling Reservoir retaining dike along North Wall	Difference	See Global Comment #9 Intent and meaning of the EALs are not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
HU4	Recognition Category: HU4	Recognition Category: HU4	Verbatim	
	Initiating Condition: FIRE potentially degrading the level of safety of the plant.	Initiating Condition: FIRE potentially degrading the level of safety of the plant.	Verbatim	
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
	(1) a. A FIRE is NOT extinguished within 15-minutes of ANY of the following FIRE detection indications: • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms or indications • Field verification of a single fire alarm **AND**	(1) a. A FIRE is NOT extinguished within 15-minutes of **ANY** of the following FIRE detection indications: • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms or indications • Field verification of a single fire alarm **AND**	Difference	See Global Comment #12
	b. The FIRE is located within ANY of the following plant rooms or areas: (site-specific list of plant rooms or areas)	b. The FIRE is located within **ANY** of the plant rooms or areas in Table H3: Table H3: Plant Rooms/Areas • Mechanical/Electrical Auxiliary Building (MEAB) • Fuel Handling Building (FHB) • Reactor Containment Building (RCB) • Essential Cooling Water Intake Structure (ECWIS) • Isolation Valve Cubicle (IVC) • Diesel Generator Building (DGB)	Differences	See Global Comment #8, #9 and #12

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
HU4	(2) a. Receipt of a single fire alarm (i.e., no other indications of a FIRE). **AND** b. The FIRE is located within ANY of the following plant rooms or areas: (site-specific list of plant rooms or areas) **AND** c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.	(2) a. Receipt of a single fire alarm (i.e., no other indications of a FIRE). **AND** b. The FIRE is located within **ANY** of the plant rooms or areas in Table H3: **AND** c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.	Difference	Syntax altered to refer to Table H3.
	(3) A FIRE within the plant or ISFSI [for plants with an ISFSI outside the plant Protected Area] PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.	(3) A FIRE within the plant PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.	Difference	STPEGS ISFSI is within the plant PROTECTED AREA; removed “or ISFSI” from EAL.
	(4) A FIRE within the plant or ISFSI [for plants with an ISFSI outside the plant Protected Area] PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.	(4) A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.	Difference	STPEGS ISFSI is within the plant PROTECTED AREA; removed “or ISFSI” from EAL. Intent and meaning of the EALs are not altered.

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
HU7	Recognition Category: HU7	Recognition Category: HU7	Verbatim	
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO) UE.	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a UE.	Difference	See Global Comment #3
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
	(1) Other conditions exist which in the judgment of the Emergency Director indicate that 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 offsite response or monitoring are expected unless further degradation of safety systems occurs.	(1) Other conditions exist which in the judgment of the Emergency Director indicate that 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 offsite response or monitoring are expected unless further degradation of safety systems occurs.	Verbatim	Intent and meaning of the EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
HA1	Recognition Category: HA1	Recognition Category: HA1	Verbatim	
	Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.	Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.	Verbatim	
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
	(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the (site-specific security shift supervision).	(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by **ANY** of the following personnel in Table H1: **Table H1: Security Supervision** • Security Force Supervisor • Acting Security Manager • Security Manager	Difference	See Global Comment #9
	(2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.	(2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.	Verbatim	Intent and meaning of the EALs are not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
HA5	Recognition Category: HA5	Recognition Category: HA5	Verbatim	
	Initiating Condition: Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown.	Initiating Condition: Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown.	Verbatim	
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
	(1) a. Release of a toxic, corrosive, asphyxiant or flammable gas into any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode applicability identified)	(1) a. Release of a toxic, corrosive, asphyxiant or flammable gas into **ANY** of the following plant rooms or areas: • Mechanical/Electrical Auxiliary Building (MEAB) • Turbine Generator Building (TGB) • Isolation Valve Cubicle (IVC) • Fuel Handling Building (FHB) • Reactor Containment Building (RCB)	Difference	See Global Comment #8 and #9
	AND b. Entry into the room or area is prohibited or impeded.	**AND** b. Entry into the room or area is prohibited or impeded.	Verbatim	Intent and meaning of the EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
HA6	Recognition Category: HA6	Recognition Category: HA6	Verbatim	
	Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations.	Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations.	Verbatim	
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
	(1) An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations).	(1) An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel (ASP).	Difference	See Global Comment #9 Intent and meaning of the EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
HA7	Recognition Category: HA7	Recognition Category: HA7	Verbatim	
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of an ALERT.	Verbatim	
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
	(1) Other conditions exist which, in the judgment of the Emergency Director, indicate that 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 Protective Action Guideline exposure levels.	(1) Other conditions exist which, in the judgment of the Emergency Director, indicate that 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 PROTECTIVE ACTION GUIDELINE exposure levels.	Difference	See Global Comment #2 Intent and meaning of the EAL is not altered.

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
HS1	Recognition Category: HS1	Recognition Category: HS1	Verbatim	
	Initiating Condition: HOSTILE ACTION within the PROTECTED AREA.	Initiating Condition: HOSTILE ACTION within the PROTECTED AREA.	Verbatim	
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
	(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision).	(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by **ANY** of the following personnel in Table H1: Table H1: Security Supervision • Security Force Supervisor • Acting Security Manager • Security Manager	Differences	See Global Comment #8 and #9 Intent and meaning of the EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
HS6	Recognition Category: HS6	Recognition Category: HS6	Verbatim	
	Initiating Condition: Inability to control a key safety function from outside the Control Room.	Initiating Condition: Inability to control a key safety function from outside the Control Room.	Verbatim	
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
	(1) a. An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations). **AND**	(1) a. An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel (ASP). **AND**	Difference	See Global Comment #9
	b. Control of **ANY** of the following key safety functions is not reestablished within (site-specific number of minutes). • Reactivity control • Core cooling [PWR] / RPV water level [BWR] • RCS heat removal	b. Control of **ANY** of the following key safety functions in Table H2 is not reestablished within 15 minutes. Table H2: Safety Functions • Reactivity control • Core cooling • RCS heat removal	Differences	See Global Comment #4, #8 and #9 Intent and meaning of the EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
HS7	Recognition Category: HS7	Recognition Category: HS7	Verbatim	
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY.	Difference	See Global Comment #2
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
	(1) Other conditions exist which in the judgment of the Emergency Director indicate that 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 Protective Action Guideline exposure levels beyond the site boundary.	(1) Other conditions exist which in the judgment of the Emergency Director indicate that 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 PROTECTIVE ACTION GUIDELINE exposure levels beyond the SITE BOUNDARY.	Verbatim	See Global Comment #2 Intent and meaning of the EAL is not altered.

<table>
<tr><th>RC</th><th>NEI 99-01 Rev. 6</th><th>South Texas Project</th><th>Change</th><th>Justification</th></tr>
<tr><td rowspan="6">HG1</td><td>Recognition Category: HG1</td><td>Recognition Category: HG1</td><td>Verbatim</td><td></td></tr>
<tr><td>Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility.</td><td>Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the FACILITY.</td><td>Difference</td><td>See Global Comment #2</td></tr>
<tr><td>Operating Mode Applicability: All</td><td>Operating Mode Applicability: ALL</td><td>Difference</td><td>See Global Comment #12</td></tr>
<tr><td>(1) a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision).</td><td>(1) a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by ANY of the following personnel in Table H1:

Table H1: Security Supervision
• Security Force Supervisor
• Acting Security Manager
• Security Manager</td><td>Differences</td><td>See Global Comment #8 and #9</td></tr>
<tr><td>AND
b. EITHER of the following has occurred:
1. ANY of the following safety functions cannot be controlled or maintained.
• Reactivity control
• Core cooling [PWR] / RPV water level [BWR]
• RCS heat removal</td><td>AND
b. EITHER of the following has occurred:
1. ANY of the following safety functions in Table H2 cannot be controlled or maintained.

Table H2: Safety Functions
• Reactivity control
• Core cooling
• RCS heat removal</td><td>Differences</td><td>See Global Comment #4, #8 and #9</td></tr>
<tr><td>OR
2. Damage to spent fuel has occurred or is IMMINENT.</td><td>OR
2. Damage to spent fuel has occurred or is IMMINENT.</td><td>Verbatim</td><td>Intent and meaning of the EAL is not altered.</td></tr>
</table>

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
HG7	Recognition Category: HG7	Recognition Category: HG7	Verbatim	
	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.	Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY	Difference	See Global Comment #2
	Operating Mode Applicability: All	Operating Mode Applicability: **ALL**	Difference	See Global Comment #12
	(1) Other conditions exist which in the judgment of the Emergency Director indicate that 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 Protective Action Guideline exposure levels offsite for more than the immediate site area.	(1) Other conditions exist which in the judgment of the Emergency Director indicate that 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 PROTECTIVE ACTION GUIDELINE exposure levels offsite for more than the immediate site area.	Difference	See Global Comment #2 Intent and meaning of the EAL is not altered.

SYSTEM MALFUNCTION ICS/EALS

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
SU1	Recognition Category: SU1	Recognition Category: SU1	Verbatim	
	Initiating Condition: Loss of all offsite AC power capability to emergency buses for 15 minutes or longer.	Initiating Condition: Loss of **ALL** offsite AC power capability to emergency buses for 15 minutes or longer.	Difference	See Global Comment #12
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	See Global Comment #6
	(1) Loss of **ALL** offsite AC power capability to (site-specific emergency buses) for 15 minutes or longer.	(1) Loss of **ALL** offsite AC power capability to **ALL** three 4160V AC ESF Busses for 15 minutes or longer.	Difference	See Global Comment #9 Intent and meaning of the EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
SU2	Recognition Category: SU2	Recognition Category: SU2	Verbatim	
	Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer.	Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer.	Verbatim	
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	See Global Comment #6
	(1) a. An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.	(1) An UNPLANNED event results in the inability to monitor one or more of the following parameters in Table S1 from within the Control Room for 15 minutes or longer:	Differences	Subheading "a" removed. Not necessary.
	[BWR parameter list] / [PWR parameter list]: Reactor Power / Reactor Power RPV Water Level / RCS Level RPV Pressure / RCS Pressure Primary Containment Pressure / In-Core/Core Exit Temperature Suppression Pool Level / Levels in at least (site-specific number) two steam generators Suppression Pool Temperature / Steam Generator Auxiliary or Emergency Feed Water Flow	Table S1: Plant Parameters • Reactor Power • RCS level • RCS Pressure • Core Exit Temperature • Levels in at least two steam generators • Steam Generator Auxiliary Feed Water Flow	Differences	See Global Comment #4, #8 and #9 Intent and meaning of the EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
SU3	Recognition Category: SU3	Recognition Category: SU3	Verbatim	
	Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits.	Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits.	Verbatim	
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	See Global Comment #6
	(1) (Site-specific radiation monitor) reading greater than (site-specific value).	(1) RT-8039 reading greater than 30 μCi/cm^3.	Difference	See Global Comment #9
	(2) Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications.	(2) Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications: • Greater than 1 μCi/gm Dose Equivalent I-131 • Greater than 100/ $\bar{E}$ μCi /gm gross activity	Difference	See Global Comment #9 Intent and meaning of the EALs are not altered.

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
SU4	Recognition Category: SU4	Recognition Category: SU4	Verbatim	
	Initiating Condition: RCS leakage for 15 minutes or longer.	Initiating Condition: RCS leakage for 15 minutes or longer.	Verbatim	
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	See Global Comment #6
	(1) RCS unidentified or pressure boundary leakage greater than (site-specific value) for 15 minutes or longer.	(1) RCS unidentified or pressure boundary leakage greater than 10 gpm for 15 minutes or longer.	Difference	See Global Comment #9
	(2) RCS identified leakage greater than (site-specific value) for 15 minutes or longer.	(2) RCS identified leakage greater than 25 gpm for 15 minutes or longer.	Difference	See Global Comment #9
	(3) Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer.	(3) Leakage from the RCS greater than 150 gallons per day through **ANY** one steam generator.	Difference	See Global Comment #9. Intent and meaning of the EALs are not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
SU5	Recognition Category: SU5	Recognition Category: SU5	Verbatim	
	Initiating Condition: Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor.	Initiating Condition: Automatic or manual trip fails to shutdown the reactor	Difference	See Global Comment #4
	Operating Mode Applicability: Power Operation	Operating Mode Applicability: 1, 2	Difference	See Global Comment #6
	(1) a. An automatic (trip [PWR] / scram [BWR]) did not shutdown the reactor. **AND**	(1) a. An automatic trip did not shutdown the reactor. **AND**	Difference	See Global Comment #4
	b. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor.	b. A subsequent manual action taken at the reactor control panels is successful in shutting down the reactor	Difference	Changed "reactor control consoles" to "reactor control panels" to reflect STP terminology.
	(2) a. A manual trip ([PWR] / scram [BWR]) did not shutdown the reactor.	(2) a. A manual trip did not shutdown the reactor.	Difference	See Global Comment #4
	AND b. EITHER of the following: 1. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor. **OR**	**AND** b. **EITHER** of the following: 1. A subsequent manual action taken at the reactor control panels is successful in shutting down the reactor. **OR**	Difference	Changed "reactor control consoles" to "reactor control panels" to reflect STP terminology.
	2. A subsequent automatic (trip [PWR] / scram [BWR]) is successful in shutting down the reactor.	2. A subsequent automatic trip is successful in shutting down the reactor.	Difference	See Global Comment #4
				Intent and mean of the IC and EALs are not altered.

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
SU6	Recognition Category: SU6	Recognition Category: SU6	Verbatim	
	Initiating Condition: Loss of all onsite or offsite communications capabilities.	Initiating Condition: Loss of all onsite or offsite communications capabilities.	Verbatim	
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	See Global Comment #6
	(1) Loss of ALL of the following Onsite communication methods: (site-specific list of communications methods)	(1) Loss of **ALL** of the following Onsite communication methods in Table S2.	Difference	See Global Comment #8 and #9
	(2) Loss of ALL of the following ORO communications methods: (site-specific list of communications methods)	(2) Loss of **ALL** of the following Offsite Response Organization communications (ORO) methods in Table S2.	Difference	See Global Comment #8 and #9
	(3) Loss of ALL of the following NRC communications methods: (site-specific list of communications methods)	(3) Loss of **ALL** of the following NRC communications methods in Table S2.	Difference	See Global Comment #8 and #9 The intent and meaning of the EALs are not altered

Table S2: Communications Methods

METHOD	EAL-1 ONSITE	EAL-2 ORO	EAL-3 NRC
• Plant PA system	X		
• Plant Radios	X		
• Plant telephone system	X	X	X
• Satellite phones		X	X
• Direct line from Control Rooms to Bay City		X	X
• Microwave Lines to Houston		X	X
• Security radio to Matagorda County		X	
• Dedicated Ring-down lines		X	
• ENS line			X

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
SU7	Recognition Category: SU7	Recognition Category: SU7	Verbatim	
	Initiating Condition: Failure to isolate containment or loss of containment pressure control. [PWR]	Initiating Condition: Failure to isolate containment or loss of containment pressure control.	Difference	See Global Comment #4
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	See Global Comment #6
	(1) a. Failure of containment to isolate when required by an actuation signal. **AND**	(1) a. Failure of containment to isolate when required by an actuation signal. **AND**	Verbatim	
	b. **ALL** required penetrations are not closed within 15 minutes of the actuation signal.	b. **ALL** required penetrations are not isolated within 15 minutes of the actuation signal.	Difference	The term "isolated" is synonymous with "closed" in STPEGS Emergency Operating Procedures implementation.
	(2) a. Containment pressure greater than (site-specific pressure). **AND**	(2) a. Containment pressure greater than 9.5 psig. **AND**	Difference	See Global Comment #9
	b. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer.	b. No Containment Spray train is operating per design for 15 minutes or longer.	Difference	Altered wording for clarification. See Global Comment #9. Intent and meaning of the EALs are not altered.

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
SA1	Recognition Category: SA1	Recognition Category: SA1	Verbatim	
	Initiating Condition: Loss of all but one AC power source to emergency buses for 15 minutes or longer.	Initiating Condition: Loss of **ALL** but one AC power source to emergency buses for 15 minutes or longer.	Difference	See Global Comment #12
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	See Global Comment #6
	(1) a. AC power capability to (site-specific emergency buses) is reduced to a single power source for 15 minutes or longer.	(1) a. AC power capability to **ALL** three 4160V AC ESF Buses is reduced to a single power source for 15 minutes or longer.	Difference	See Global Comment #9 and #12
	AND	**AND**		
	b. Any additional single power source failure will result in a loss of all AC power to SAFETY SYSTEMS.	b. **ANY** additional single power source failure will result in a loss of **ALL** AC power to SAFETY SYSTEMS.	Difference	See Global Comment #12 Intent and meaning of the EAL is not altered.

<table>
<tr><th>RC</th><th>NEI 99-01 Rev. 6</th><th>South Texas Project</th><th>Change</th><th>Justification</th></tr>
<tr><td rowspan="6">SA2</td><td>Recognition Category: SA2</td><td>Recognition Category: SA2</td><td>Verbatim</td><td></td></tr>
<tr><td>Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.</td><td>Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.</td><td>Verbatim</td><td></td></tr>
<tr><td>Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown</td><td>Operating Mode Applicability: 1, 2, 3, 4</td><td>Differences</td><td>See Global Comment #6</td></tr>
<tr><td>(1) a. An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.</td><td>(1) a. An UNPLANNED event results in the inability to monitor one or more of the following parameters in Table S1 from within the Control Room for 15 minutes or longer.</td><td>Difference</td><td>See Global Comment #8 and #9</td></tr>
<tr><td><table><tr><th>[BWR parameter list]</th><th>[PWR parameter list]</th></tr><tr><td>Reactor Power</td><td>Reactor Power</td></tr><tr><td>RPV Water Level</td><td>RCS Level</td></tr><tr><td>RPV Pressure</td><td>RCS Pressure</td></tr><tr><td>Primary Containment Pressure</td><td>In-Core/Core Exit Temperature</td></tr><tr><td>Suppression Pool Level</td><td>Levels in at least (site-specific number) steam generators</td></tr><tr><td>Suppression Pool Temperature</td><td>Steam Generator Auxiliary or Emergency Feed Water Flow</td></tr></table>
AND</td><td><table><tr><th>Table S1: Plant Parameters</th></tr><tr><td>• Reactor Power
• RCS level
• RCS Pressure
• Core Exit Temperature
• Levels in at least two steam generators
• Steam Generator Auxiliary Feed Water Flow</td></tr></table>
AND</td><td>Differences</td><td>See Global Comment #4</td></tr>
<tr><td>b. ANY of the following transient events in progress.
• Automatic or manual runback greater than 25% thermal reactor power
• Electrical load rejection greater than 25% full electrical load
• Reactor scram [BWR] / trip [PWR]
• ECCS (SI) actuation
• Thermal power oscillations greater than 10% [BWR]</td><td>b. ANY of the following transient events in progress.
• Automatic or manual runback greater than 25% thermal reactor power
• Electrical load rejection greater than 25% full electrical load
• Reactor trip
• ECCS (SI) actuation</td><td>Differences</td><td>See Global Comment #4

Intent and meaning of the EAL is not altered.</td></tr>
</table>

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
SA5	Recognition Category: SA5	Recognition Category: SA5	Verbatim	
	Initiating Condition: Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	Initiating Condition: Automatic or manual trip fails to shutdown the reactor, and subsequent manual actions taken at the reactor control panels are not successful in shutting down the reactor.	Differences	See Global Comment #4 Changed "consoles" to "panels" to reflect STPEGS terminology.
	Operating Mode Applicability: Power Operation	Operating Mode Applicability: 1, 2	Difference	See Global Comment #6
	(1) a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. Manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	(1) a. An automatic or manual trip did not shutdown the reactor. **AND** b. Manual actions taken at the reactor control panels are not successful in shutting down the reactor.	Difference Difference	See Global Comment #4 Changed "consoles" to "panels" to reflect STP terminology. Intent and meaning of the EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
SA9	Recognition Category: SA9	Recognition Category: SA9	Verbatim	
	Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.	Verbatim	
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	See Global Comment #6
	(1) a. The occurrence of ANY of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • (site-specific hazards) • Other events with similar hazard characteristics as determined by the Shift Manager **AND**	(1) a. The occurrence of **ANY** of the following hazardous events listed in Table S3: Table S3: Hazardous Events • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Predicted or actual breach of Main Cooling Reservoir retaining dike along North Wall. • Other events with similar hazard characteristics as determined by the Shift Manager **AND**	Differences	See Global Comment #8, #9 and #12
	b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode. **OR**	b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode. **OR**	Verbatim	
	2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	Verbatim	Intent and meaning of the EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
SS1	Recognition Category: SS1	Recognition Category: SS1	Verbatim	
	Initiating Condition: Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.	Initiating Condition: Loss of **ALL** offsite and **ALL** onsite AC power to emergency buses for 15 minutes or longer.	Difference	See Global Comment #12
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	See Global Comment #6
	(1) Loss of ALL offsite and ALL onsite AC power to (site-specific emergency buses) for 15 minutes or longer.	(1) Loss of **ALL** offsite and **ALL** onsite AC power to **ALL** three 4160V AC ESF Buses for 15 minutes or longer.	Differences	See Global Comment #9 and #12 Intent and meaning of the EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
SS5	Recognition Category: SS5	Recognition Category: SS5	Verbatim	
	Initiating Condition: Inability to shutdown the reactor causing a challenge to (core cooling [PWR] / RPV water level [BWR]) or RCS heat removal.	Initiating Condition: Inability to shutdown the reactor causing a challenge to core cooling or RCS heat removal.	Difference	See Global Comment #4
	Operating Mode Applicability: Power Operation	Operating Mode Applicability: 1, 2	Difference	See Global Comment #6
	(1) a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor.	(1) a. An automatic or manual trip did not shutdown the reactor.	Difference	See Global Comment #4
	AND b. All manual actions to shutdown the reactor have been unsuccessful.	**AND** b. **ALL** manual actions to shutdown the reactor have been unsuccessful.	Difference	See Global Comment #12
	AND c. EITHER of the following conditions exist: • (Site-specific indication of an inability to adequately remove heat from the core) • (Site-specific indication of an inability to adequately remove heat from the RCS)	**AND** c. **EITHER** of the following conditions exist: • Core Cooling – Red entry condition met • Heat Sink- Red entry condition met	Differences	See Global Comment #9 Intent and meaning of the EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
SS8	Recognition Category: SS8	Recognition Category: SS8	Verbatim	
	Initiating Condition: Loss of all Vital DC power for 15 minutes or longer.	Initiating Condition: Loss of **ALL** Vital DC power for 15 minutes or longer.	Difference	See Global Comment #12
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	See Global Comment #6
	(1) Indicated voltage is less than (site-specific bus voltage value) on **ALL** (site-specific Vital DC busses) for 15 minutes or longer.	(1) Indicated voltage is less than 105.5 VDC on **ALL** Class 1E 125 VDC battery buses for 15 minutes or longer.	Difference	See Global Comment #9 Intent and meaning of the EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
SG1	Recognition Category: SG1	Recognition Category: SG1	Verbatim	
	Initiating Condition: Prolonged loss of all offsite and all onsite AC power to emergency buses.	Initiating Condition: Prolonged loss of **ALL** offsite and **ALL** onsite AC power to emergency buses.	Difference	See Global Comment #12
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	See Global Comment #6
	(1) a. Loss of ALL offsite and ALL onsite AC power to (site-specific emergency buses). **AND** b. EITHER of the following:	(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to **ALL** three 4160V AC ESF Buses **AND** b. **EITHER** of the following:	Difference	See Global Comment #9 and #12
	• Restoration of at least one AC emergency bus in less than (site-specific hours) is not likely.	• Restoration of at least one 4160V AC ESF bus in less than 4 hours is not likely.	Difference	See Global Comment #9
	• (Site-specific indication of an inability to adequately remove heat from the core)	• Core Cooling – Red entry condition met	Difference	See Global Comment #9 Intent and meaning of EAL is not altered.

RC	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
SG8	Recognition Category: SG8	Recognition Category: SG8	Verbatim	
	Initiating Condition: Loss of all AC and Vital DC power sources for 15 minutes or longer.	Initiating Condition: Loss of **ALL** AC and Vital DC power sources for 15 minutes or longer.	Difference	See Global Comment #12
	Operating Mode Applicability: Power Operation, Startup, Hot Standby, Hot Shutdown	Operating Mode Applicability: 1, 2, 3, 4	Difference	See Global Comment #6
	(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to (site-specific emergency buses) for 15 minutes or longer. **AND**	(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to **ALL** three 4160 V AC ESF buses for 15 minutes or longer **AND**	Difference	See Global Comment #9 and #12
	b. Indicated voltage is less than (site-specific bus voltage value) on ALL (site-specific Vital DC busses) for 15 minutes or longer.	b. Indicated voltage is less than 105.5 VDC on **ALL** Class 1E 125 VDC battery buses for 15 minutes or longer.	Difference	See Global Comment #9 Intent and meaning of EAL is not altered.

APPENDIX A – ACRONYMS AND ABBREVIATIONS

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

Section	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
APPENDIX A – ACRONYMS AND ABBREVIATIONS	AC.......Alternating Current	AC.......Alternating Current	Verbatim	
	AOP......Abnormal Operating Procedure	AOP......Abnormal Operating Procedure	Verbatim	
	APRM...Average Power Range Meter		Deleted	Not Applicable
	ATWS...Anticipated Transient Without Scram	ATWS...Anticipated Transient Without Scram	Verbatim	
	B&W....Babcock and Wilcox		Deleted	Not Applicable
	BIIT......Boron Injection Initiating Temperature		Deleted	Not Applicable
	BWR....Boiling Water Reactor		Deleted	Not Applicable
	CDE......Committed Dose Equivalent	CDE......Committed Dose Equivalent	Verbatim	
	CFR......Code of Federal Regulations	CFR......Code of Federal Regulations	Verbatim	
	CTMT/CNMT...Containment	CTMT/CNMT...Containment	Verbatim	
	CSF......Critical Safety Function	CSF......Critical Safety Function	Verbatim	
	CSFST...Critical Safety Function Status Tree	CSFST...Critical Safety Function Status Tree	Verbatim	
	DBA......Design Basis Accident	DBA......Design Basis Accident	Verbatim	
	DC........Direct Current	DC........Direct Current	Verbatim	
	EAL.......Emergency Action Level	EAL.......Emergency Action Level	Verbatim	
	ECCS....Emergency Core Cooling System	ECCS....Emergency Core Cooling System	Verbatim	
	ECL.......Emergency Classification Level	ECL.......Emergency Classification Level	Verbatim	
	EOF......Emergency Operations Facility	EOF......Emergency Operations Facility	Verbatim	
	EOP......Emergency Operating Procedure	EOP......Emergency Operating Procedure	Verbatim	
	EPA......Environmental Protection Agency	EPA......Environmental Protection Agency	Verbatim	
	EPG.....Emergency Procedure Guideline	EPG.....Emergency Procedure Guideline	Verbatim	
	EPIP.....Emergency Planning Implementing Procedure		Deleted	Not Applicable
	EPR......Evolutionary Power Reactor		Deleted	Not Applicable
	EPRI.....Electric Power Research Institute		Deleted	Not Applicable
	ERG.....Emergency Response Guideline	ERG.....Emergency Response Guideline	Verbatim	
	FEMA...Federal Emergency Management Agency	FEMA...Federal Emergency Management Agency	Verbatim	
	FSAR....Final Safety Analysis Report	FSAR....Final Safety Analysis Report	Verbatim	
	GE........General Emergency	GE........General Emergency	Verbatim	
	HCTL....Heat Capacity Temperature Limit		Deleted	Not Applicable
	HPCI.....High Pressure Coolant Injection		Deleted	Not Applicable
	HSI........Human System Interface		Deleted	Not Applicable
	IC.........Initiating Condition	IC.........Initiating Condition	Verbatim	

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

Section	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
APPENDIX A – ACRONYMS AND ABBREVIATIONS	ID........Inside Diameter	ID........Inside Diameter	Verbatim	
	IPEEE...Individual Plant Examination of External Events (Generic Letter 88-20)		Deleted	Not Applicable
	ISFSI....Independent Spent Fuel Storage Installation	ISFSI....Independent Spent Fuel Storage Installation	Verbatim	
	Keff.....Effective Neutron Multiplication Factor	Keff.....Effective Neutron Multiplication Factor	Verbatim	
	LCO.....Limited Condition of Operation	LCO.....Limited Condition of Operation	Verbatim	
	LOCA...Loss of Coolant Accident	LOCA...Loss of Coolant Accident	Verbatim	
	MCR....Main Control Room		Verbatim	
	MSIV...Main Steam Isolation Valve	MSIV...Main Steam Isolation Valve	Verbatim	
	MSL.....Main Stem Line	MSL.....Main Stem Line	Verbatim	
	mR, mRem, mrem, mREM....milli-Roentgen Equivalent Man	mR, mRem, mrem, mREM....milli-Roentgen Equivalent Man	Verbatim	
	MW.....Megawatt	MW.....Megawatt	Verbatim	
	NEI.......Nuclear Energy Institute	NEI.......Nuclear Energy Institute	Verbatim	
	NPP......Nuclear Power Plant	NPP......Nuclear Power Plant	Verbatim	
	NRC.....Nuclear Regulatory Agency	NRC.....Nuclear Regulatory Agency	Verbatim	
	NSSS....Nuclear Steam Supply System	NSSS....Nuclear Steam Supply System	Verbatim	
	NORAD...North American Aerospace Defense Command	NORAD...North American Aerospace Defense Command	Verbatim	
	(NO)UE...(Notification of) Unusual Event	UE........Unusual Event	Difference	STP uses UE
	NUMARC....Nuclear Management and Resources Council	NUMARC....Nuclear Management and Resources Council	Verbatim	
	OBE.....Operating Basis Earthquake	OBE.....Operating Basis Earthquake	Verbatim	
	OCA.....Owner Controlled Area	OCA.....Owner Controlled Area	Verbatim	
	ODCM/ODAM....Offsite Dose Calculation (Assessment) Manual	ODCM...Offsite Dose Calculation Manual	Difference	STP uses ODCM
	ORO.....Offsite Response Organization	ORO.....Offsite Response Organization	Verbatim	
	PA........Protected Area	PA........Protected Area	Verbatim	
	PACS....Priority Information and Control System		Deleted	Not Applicable
	PAG......Protective Action Guideline	PAG......Protective Action Guideline		
	PICS.....Process Information and Control System		Deleted	Not Applicable
	PRA/PSA...Probabilistic Risk Assessment/Probabilistic Safety Assessment	PRA/PSA...Probabilistic Risk Assessment/Probabilistic Safety Assessment	Verbatim	
	PWR....Pressurized Water Reactor	PWR....Pressurized Water Reactor	Verbatim	

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

Section	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
APPENDIX A – ACRONYMS AND ABBREVIATIONS	PS.........Protection System		Deleted	Not Applicable
	PSIG.....Pounds per Square Inch	PSIG.....Pounds per Square Inch	Verbatim	
	R..........Roentgen	R..........Roentgen	Verbatim	
	RCC....Reactor Control Console		Deleted	Not Applicable
	RCIC...Reactor Core Isolation Cooling		Deleted	Not Applicable
	RCS.....Reactor Coolant System	RCS.....Reactor Coolant System	Verbatim	
	Rem, rem, REM...Roentgen Equivalent Man	Rem, rem, REM...Roentgen Equivalent Man	Verbatim	
	RETS....Radiological Effluent Technical Specifications		Deleted	Not Applicable
	RPS......Reactor Protection System	RPS......Reactor Protection System	Verbatim	
	RPV......Reactor Pressure Vessel	RPV......Reactor Pressure Vessel	Verbatim	
	RVLIS...Reactor Vessel Level Instrumentation System	RVWL...Reactor Vessel Water Level	Difference	STP uses RVWL
	RWCU...Reactor Water Cleanup		Deleted	Not Applicable
	SAR.......Safety Analysis Report	SAR.......Safety Analysis Report	Verbatim	
	SAS........Safety Automation System		Deleted	Not Applicable
	SBO.......Station Blackout		Deleted	Not Applicable
	SCBA.....Self-Contained Breathing Apparatus	SCBA.....Self-Contained Breathing Apparatus	Verbatim	
	SG..........Steam Generator	SG..........Steam Generator	Verbatim	
	SI...........Safety Injection	SI...........Safety Injection	Verbatim	
	SICS......Safety Information Control System		Deleted	Not Applicable
	SPDS.....Safety Parameter Display System	SPDS.....Safety Parameter Display System	Verbatim	
	SRO.......Senior Reactor Operator		Deleted	Not Applicable
	TEDE.....Total Effective Dose Equivalent	TEDE.....Total Effective Dose Equivalent	Verbatim	
	TOAF.....Top of Active Fuel	TOAF.....Top of Active Fuel	Verbatim	
	TSC........Technical Support System	TSC........Technical Support System	Verbatim	
	WOG.....Westinghouse Owners Group	WOG.....Westinghouse Owners Group	Verbatim	

APPENDIX B - DEFINITIONS

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

Section	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
APPENDIX B - DEFINITIONS	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.	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.	Difference	See Global Comment #2
	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 offsite for more than the immediate site area.	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 offsite for more than the immediate site area.	Difference	See Global Comment #2
	Notification of 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 offsite response or monitoring are expected unless further degradation of safety systems occurs.	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 offsite response or monitoring are expected unless further degradation of safety systems occurs.	Difference	See Global Comment #2
	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.	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.	Difference	See Global Comment #2
	Emergency Action Level (EAL): A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.	EMERGENCY ACTION LEVEL (EAL): A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.	Difference	See Global Comment #2

Section	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
APPENDIX B - DEFINITIONS	Emergency Classification Level (ECL): One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are: • UNUSUAL EVENT (UE) • ALERT • SITE AREA EMERGENCY (SAE) • GENERAL EMERGENCY (GE)	EMERGENCY CLASSIFICATION LEVEL (ECL): One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are: • UNUSUAL EVENT (UE) • ALERT • SITE AREA EMERGENCY (SAE) • GENERAL EMERGENCY (GE)	Difference	See Global Comment #2
	Fission Product Barrier Threshold: A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.	FISSION PRODUCT BARRIER THRESHOLD: A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.	Difference	See Global Comment #2
	Initiating Condition (IC): An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.	INITIATING CONDITION (IC): An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.	Difference	See Global Comment #2
	CONFINEMENT BOUNDARY: (Insert a site-specific definition for this term.) **Developer Note** – The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.	CONFINEMENT BOUNDARY: The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.	Difference	Removed developer notes.
	CONTAINMENT CLOSURE: (Insert a site-specific definition for this term.) **Developer Note** – The procedurally defined conditions or actions taken to secure containment (primary or secondary for BWR) and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.	CONTAINMENT CLOSURE: The procedurally defined conditions or actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.	Difference	Removed developer notes.
		CREDIBLE SECURITY THREAT: Information received from a source determined to be reliable (e.g., law enforcement, government agency, etc.) or has been verified to be true or considered credible when: (1) Physical evidence supporting the threat exists, (2) Information independent from the actual threat message exists that supports the threat, or (3) A specific known group or organization claims responsibility for the threat.	Difference	Added definition. Term is defined in Security Plan.

Section	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
APPENDIX B - DEFINITIONS	EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.	EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.	Verbatim	
		FACILITY: The area and buildings within the PROTECTED AREA and the switchyard.	Difference	Added definition. Term is defined in Security Plan
	FAULTED: The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized. Developer Note – This term is applicable to PWRs only.	FAULTED: The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.	Difference	Removed developer notes.
	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.	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.	Verbatim	
		HATCH MONITOR: Temporary monitor installed when Containment High Range Radiation Monitors RT-8050 and RT-8051 are out of service.	Difference	Added definition. Monitor used in EALs.
	HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.	HOSTAGE: A person(s) held as leverage against the station to ensure that demands will be met by the station.	Verbatim	
	HOSTILE ACTION: An act toward a NPP 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 the NPP. 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).	HOSTILE ACTION: An act toward a NPP 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 the NPP. 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).	Verbatim	

Section	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
APPENDIX B - DEFINITIONS	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.	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.	Verbatim	
	IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.	IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.	Verbatim	
	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.	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.	Verbatim	
	NORMAL LEVELS: As applied to radiological IC/EALs, the highest reading in the past twenty-four hours excluding the current peak value.	NORMAL LEVELS: As applied to radiological IC/EALs, the highest reading in the past twenty-four hours excluding the current peak value.	Verbatim	
	OWNER CONTROLLED AREA: (Insert a site-specific definition for this term.) **Developer Note** – This term is typically taken to mean the site property owned by, or otherwise under the control of, the licensee. In some cases, it may be appropriate for a licensee to define a smaller area with a perimeter closer to the plant Protected Area perimeter (e.g., a site with a large OCA where some portions of the boundary may be a significant distance from the Protected Area). In these cases, developers should consider using the boundary defined by the Restricted or Secured Owner Controlled Area (ROCA/SOCA). The area and boundary selected for scheme use must be consistent with the description of the same area and boundary contained in the Security Plan.	OWNER CONTROLLED AREA: The area surrounding the PROTECTED AREA where STP Nuclear Operating Company (STPNOC) reserves the right to restrict access, search personnel, and vehicles.	Difference	Site-specific definition consistent with the Security Plan used.
	PROJECTILE: An object directed toward a NPP that could cause concern for its continued operability, reliability, or personnel safety.	PROJECTILE: An object directed toward a NPP that could cause concern for its continued operability, reliability, or personnel safety.	Verbatim	
	PROTECTED AREA: (Insert a site-specific definition for this term.) **Developer Note** – This term is typically taken to mean the area under continuous access monitoring and control, and armed protection as described in the site Security Plan.	PROTECTED AREA: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.	Difference	Definition from developer notes used.
		PROTECTIVE ACTION GUIDES (PAG): Environmental Protection Agency (EPA) guides for protective actions to safeguard against radiation exposure from nuclear incidents.	Difference	Added definition. Used in EALS.

Section	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
APPENDIX B - DEFINITIONS	REFUELING PATHWAY: (Insert a site-specific definition for this term.) **Developer Note** – This description should include all the cavities, tubes, canals and pools through which irradiated fuel may be moved, but not including the reactor vessel.	REFUELING PATHWAY: Includes all the cavities, tubes, canals and pools through which irradiated fuel may be moved, but not including the reactor vessel.	Difference	Definition from developer notes used.
	RUPTURE(D): The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection. **Developer Note** – This term is applicable to PWRs only.	RUPTURE(D): The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.	Difference	Removed developer notes.
	SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related. **Developer Note** – This term may be modified to include the attributes of "safety-related" in accordance with 10 CFR 50.2 or other site-specific terminology, if desired.	SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related. **Developer Note** – This term may be modified to include the attributes of "safety-related" in accordance with 10 CFR 50.2 or other site-specific terminology, if desired.	Difference	Removed developer notes.
	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.	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.	Verbatim	
		SECURITY EVENT: Any incident representing an attempted, threatened, of actual breach of the security system of reduction of the operational effectiveness of that system. A security event can result in either a SECURITY CONDITION or HOSTILE ACTION.	Difference	Added definition. Term in defined in Security Plan.
		SITE BOUNDARY: The edge of the plant property whose access may be controlled by STPEGS. This boundary is congruent with the Exclusion Area Boundary for the purpose of offsite dose assessment.	Difference	Added definition. Used as site-specific dose receptor point in EALs.
		THYROID CDE: The dose equivalent to the thyroid from an intake of radioactive material by an individual during the 50-year period following the intake.	Difference	Added definition from 10 CFR 20.1003.
	UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.	UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.	Verbatim	

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

Section	NEI 99-01 Rev. 6	South Texas Project	Change	Justification
APPENDIX B - DEFINITIONS	UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.	UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.	Verbatim	
		VALID: An indication, report or condition is considered to be VALID when it is verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy. This may be accomplished through an instrument channel check, response on related or redundant indicators, or direct observation by plant personnel. The verification methods should be completed in a manner the supports timely emergency declaration.	Difference	Added definition per NEI text in Section 5.1.
	VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.	VISIBLE DAMAGE: Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.	Verbatim	

ATTACHMENT 4
Supporting Documents (1548 pages)

RU1

ENERCON Excellence—Every project. Every day	CALCULATION COVER SHEET	CALC. NO.STPNOC013-CALC-002 REV. 1 PAGE NO. 1 of 49

Title:	Radiological Release Thresholds for Emergency Action Levels	**Client:** South Texas Project **Project:** STPNOC013

Item	Cover Sheet Items	Yes	No
1	Does this calculation contain any open assumptions that require confirmation? (If **YES**, Identify the assumptions) ____________		✓
2	Does this calculation serve as an "Alternate Calculation"? (If **YES**, Identify the design verified Calculation.) **Design Verified Calculation No.** ____________		✓
3	Does this calculation Supersede an existing Calculation? (If **YES**, identify the superseded Calculation.) **Superseded Calculation No.** ____________		✓

Scope of Revision: Incorporate decay time of one hour from shutdown as well as migration into Attachment 1. Change statement of no decay in the STAMPEDE runs.

Revision Impact on Results: Values calculated in Attachment 1 decreased and have become the limiting values.

~~**Study Calculation**~~ ☐ ~~**Final Calculation**~~ ☑

Safety-Related ☑ **Non-Safety Related** ☐

(Print Name and Sign)

Originator:	Caleb Trainor	**Date:** 3/21/2014
Design Verifier:	Chad Cramer	**Date:** 3/21/14
Approver:	Marvin Morris	**Date:** 3/21/14

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
0	03/03/2014	Initial Issue
1	3/21/2014	Resolve inconsistency in decay times for the two calculations

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION
1-11	1		

ATTACHMENT REVISION STATUS

ATTACHMENT NO.	PAGE NO.	REVISION NO.	ATTACHMENT NO.	PAGE NO.	REVISION NO.
1	12-24	1			
2	25-31	1			
3	32-49	1			

ENERCON Excellence—Every project. Every day	CALCULATION DESIGN VERIFICATION CHECKLIST	CALC. NO. STPNOC013-CALC-002 REV. 1 PAGE NO. 3 of 49

Item	CHECKLIST ITEMS	Yes	No	N/A
1	**Design Inputs** - Were the design inputs correctly selected, referenced (latest revision), consistent with the design basis and incorporated in the calculation?	✓		
2	**Assumptions** – Were the assumptions reasonable and adequately described, justified and/or verified, and documented?	✓		
3	**Quality Assurance** – Were the appropriate QA classification and requirements assigned to the calculation?	✓		
4	**Codes, Standard and Regulatory Requirements** – Were the applicable codes, standards and regulatory requirements, including issue and addenda, properly identified and their requirements satisfied?	✓		
5	**Construction and Operating Experience** – Have applicable construction and operating experience been considered?			✓
6	**Interfaces** – Have the design interface requirements been satisfied, including interactions with other calculations?	✓		
7	**Methods** – Was the calculation methodology appropriate and properly applied to satisfy the calculation objective?	✓		
8	**Design Outputs** – Was the conclusion of the calculation clearly stated, did it correspond directly with the objectives and are the results reasonable compared to the inputs?	✓		
9	**Radiation Exposure** – Has the calculation properly considered radiation exposure to the public and plant personnel?	✓		
10	**Acceptance Criteria** – Are the acceptance criteria incorporated in the calculation sufficient to allow verification that the design requirements have been satisfactorily accomplished?	✓		
11	**Computer Software** – Is a computer program or software used, and if so, are the requirements of CSP 3.02 met?			✓

COMMENTS:
None

	(Print Name and Sign)		
Design Verifier:	Chad Cramer [signature]	**Date:**	3/21/14
Others:		**Date:**	

ENERCON Excellence—Every project. Every day	CALCULATION DESIGN VERIFICATION PLAN AND SUMMARY SHEET	CALC. NO. STPNOC013-CALC-002
		REV. 1
		PAGE NO. 4 of 49

Calculation Design Verification Plan:

Calculation shall be verified by comparing the documented input with the references and checking the validity of the references for the intended use. As necessary, assumptions shall be evaluated and verified to determine if they are based on sound engineering principles and practices. Verify the applicable methodology, inputs, results, and conclusions.

(Print Name and Sign for Approval – mark "N/A" if not required)

Approver: Marvin Morris Marvin Morris	**Date** 3/21/14

Calculation Design Verification Summary:

Design inputs, assumptions, methodology, results and conclusions were evaluated/verified and found to be acceptable. All comments have been incorporated.

Based On The Above Summary, The Calculation Is Determined To Be Acceptable.

(Print Name and Sign)

Design Verifier: Chad Cramer	**Date:** 3/21/14
Others:	**Date:**

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1.0 OBJECTIVE/SCOPE

The purpose of this calculation is to determine the Emergency Action Level (EAL) threshold values of a radiological release from the Unit Vent or Main Steam Lines for an Unusual Event, Alert, Site Area Emergency, or General Emergency. The calculated threshold values are to be included in the STP EAL Technical Basis document, which implements the new NEI 99-01, Revision 6, Emergency Action Level Scheme and will be submitted to the NRC for approval. Upon NRC approval, the values will be used in 0ERP01-ZV-IN01, Revision 10, Emergency Classification.

Both a hand calculation and the South Texas Assessment Model Projecting Emergency Dose Evaluation (STAMPEDE) software program were used to generate the results. The hand calculation is included as Attachment 1.

Revision 1 of this calculation incorporated decay for a release taking place one hour after reactor shutdown. This was done to create continuity between the two methodologies present.

2.0 SUMMARY OF RESULTS

The results of the calculations for the radiation monitors specified in the STP EAL Basis Document and are listed in Table 2.1, below.

Table 2.1: Summary of Calculation Results

Emergency Action Level		**RT-8010B, Unit Vent (µCi/sec)**	**RT-8046 through 8049, Main Steam Lines (µCi/cc)**
RU1	**Unusual Event**		
	Hand Calculation	1.40E+05	5.00E-02
	STAMPEDE	N/A*	N/A*
RA1	**Alert**		
	Hand Calculation	1.57E+06	4.10 E+00
	STAMPEDE	2.50E+06	4.50E+00
RS1	**Site Area Emergency**		
	Hand Calculation	1.57E+07	4.105E+01
	STAMPEDE	2.50E+07	4.50E+01
RG1	**General Emergency**		
	Hand Calculation	1.57E+08	4.10E+02
	STAMPEDE	2.50E+08	4.50E+02

*STAMPEDE was not used to determine the threshold for RU1. Reference 5.10 indicates that the ODCM methodology should be used to determine the threshold value.

This calculation will be used to establish the threshold values for abnormal radiation based emergencies in the STP EAL Technical Basis document.

3.0 METHOD OF ANALYSIS

Previously, STAMPEDE was used to calculate the Emergency Action Level threshold values for effluent releases. A hand calculation will verify the STAMPEDE calculations. The hand calculation is described in Attachment 1 of this document STAMPEDE conforms to the requirements of STP Procedure 0PGP07-ZA-0014, Software Quality Assurance Program. STAMPEDE was run at STP on an STP computer and under the supervision of an ENERCON employee with access to the STP site as a critical worker.

4.0 INPUTS

4.1 Per NEI 99-01, Revision 6, Initiating condition AU1, EAL 1, the Notice of Unusual Event initiating condition is a release of gaseous or liquid radioactivity greater than two times the ODCM limit for sixty minutes or longer (Reference 5.10).

4.2 The ODCM offsite dose limit is exceeded if the Xe-133 release concentration exceeds 7.41E-04 μCi/cc (Reference 5.6).

4.3 The Unit Vent flow rate is 9.4E+07 cc/sec (Reference 5.1).

4.4 The main steam line pressure and PORV choke flow rate are 1285 psig and 1.05E+06 lbm/hr, respectively (Reference 5.2).

4.5 The specific volume of saturated steam at 1285 psig is 0.338 ft^3/lbm (Reference 5.3).

4.6 The release concentration is varied to find the release concentration which correlates to each emergency action level. Emergency action levels are taken from NEI 99-01, Revision 6 (Reference 5.10) for initiating conditions AA1, AS1 and AG1. EAL 1 is the EAL of interest in each initiating condition. The doses at the Site Boundary that correlate to the threshold concentrations are listed in Table 4.1.

Table 4.1 EAL Offsite Dose Initiating Conditions

	Alert	Site Area	General
TEDE	10 mrem	100 mrem	1000 mrem
Thyroid CDE	50 mrem	500 mrem	5000 mrem

5.0 REFERENCES

5.1 Offsite Dose Calculation Manual, Revision 17, March 2011
5.2 Main Steam PORV Capacity Verification MC05591, Revision: 1
5.3 NIST Steam Tables, 2011
5.4 0ERP01-ZV-IN01, Emergency Classification Draft Revision 10
5.5 0ERP01-ZV-TP01, Offsite Dose Calculations, Revision 21
5.6 STP Calculation NC-9012, CRMS Rad Monitor Setpoints, Revision 7
5.7 STP Calculation NC-9011, Revision 2
5.8 STAMPEDE Computer Program, Revision 7.0.3.3
5.9 STAMPEDE User's Manual
5.10 NEI 99-01, Revision 6, Development of Emergency Action Levels for Non-Passive Reactors
5.11 0PGP07-ZA-0014 Quality Assurance Program
5.12 ITWMS Call Number 1000010987 Design Document, Revision 0

6.0 ASSUMPTIONS

6.1 Unit Vent Noble Gas Monitor

To be consistent with the ODCM methodology, the unit vent release is assumed to be entirely Xe-133. The unit vent noble gas monitor is calibrated to Xe-133 (Reference 5.1) therefore; the monitor reading accurately reflects the Xe-133 release magnitude.

To be consistent with ODCM methodology, the main steam line release is assumed to be entirely Xe-133. The noble gas monitor is calibrated to Xe-133 (Reference 5.6).

6.2 Release Duration

Per Reference 5.10, Sections IC AA1, AS1, and AG1 developer notes, the release should be assumed to last one hour.

6.3 Release following Reactor Shutdown

The release initiates one hour after reactor shutdown. While a release initiating at reactor shutdown is likely, significant decay of short lived nuclides occurs during the migration time. A release at reactor shutdown would have a significantly higher activity at the monitor location than at the reception site. It is important for the threshold to not be calculated at shutdown as this would create a very high threshold which would not be appropriate for releases which occur shortly after shutdown. One hour after reactor shutdown is sufficient time to decay short lived nuclides and create a conservative threshold.

6.4 Source Term

Per Reference 5.1, any unit vent release with increased RCS activity and no core melt should be calculated using the gap inventory. Therefore, the gap inventory is used for all unit vent releases.

Per Reference 5.1, for a main steam line release following a steam generator tube rupture it is appropriate to use an inventory of noble gases plus 0.2% iodine. A steam generator tube rupture is the only scenario which would create significant offsite doses through a main steam line release.

6.5 Default STAMPEDE Input Values

Reference 5.10 developer notes for initiating conditions AA1, AS1 and AG1 suggest using the ODCM or the site's emergency dose assessment methodology. STAMPEDE is used for emergency dose assessment. Per Reference 5.1, when actual meteorology is not available, the default STAMPEDE values should be used. Had the ODCM methodology been used, the 500 hour peak χ/Q value would be used which is less conservative than the χ/Q value produced by STAMPEDE using default meteorological conditions. Therefore, the use of STAMPEDE default values provides a more conservative estimate than that of the alternative method outlined in Reference 5.10.

6.6 Average Effluent Concentration (χ/Q)

The same χ/Q is used for the unit vent and main steam line release. Reference 5.1 applies the same unit vent χ/Q to Units 1 and 2 which would also be applicable to the main steam line. All releases are considered to be ground level releases.

7.0 STAMPEDE CALCULATIONS

7.1 Unusual Event – RU1

7.1.1 Unit Vent Monitor

AU1 recommends declaring an unusual event due to a release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer (Reference 5.10).

STP sets the ODCM limit at 7.41E-04 μCi/cc (Reference 5.6, pg. 16). Two times the limit would be 1.48E-03 μCi/cc. The threshold is listed in μCi/sec so that variations in flow rate do not change the threshold. The normal flow rate from the unit vent is 9.4E+07 cc/sec (Reference 5.1).

$$Concentration\left(\frac{\mu Ci}{cc}\right) * Flow\ Rate\left(\frac{cc}{sec}\right) = Release\ Rate\left(\frac{\mu Ci}{sec}\right)$$

$$(1.48E-03)\left(\frac{\mu Ci}{cc}\right) * (9.4\text{E}+07)\left(\frac{cc}{sec}\right) = 1.4E+05\left(\frac{\mu Ci}{sec}\right)$$

Equation 7.1.1.1

7.1.2 Main Steam Line Monitor

The ODCM does not calculate a release corresponding to allowable limits for the main steam line monitors. Since the unit vent release calculated in the ODCM was assumed to be primarily Xe-133, the assumption is made in the ODCM that other noble gases and iodine may be ignored in the calculation. This assumption is equally justifiable for the main steam line and the same limiting release will be used.

The magnitude of the release calculated for the unit vent Unusual Event applies to the main steam lines as well. The main steam line PORV's will create a dose exceeding two times the ODCM limit by releasing 1.4E+05 μCi/sec of activity which is equivalent to the release from the unit vent.

The steam lines hold saturated steam at 1285 psig, per Reference 5.2, which has a specific volume of 0.338 ft^3/lbm (Reference 5.3). The PORVs will release the steam at 1.05E+06 lbm/hr per Reference 5.2. This creates a set flow rate of steam from the main steam lines of 2.79E+06 cc/sec as shown below.

$$F\left(\frac{lbm}{hr}\right) * Density\left(\frac{ft^3}{lbm}\right) * 28316.846\left(\frac{cc}{ft^3}\right) \div 3600\left(\frac{sec}{hr}\right) = \frac{cc}{sec}$$

$$1.05E+06\left(\frac{lbm}{hr}\right) * 0.338\left(\frac{ft^3}{lbm}\right) * 28316.846\left(\frac{cc}{ft^3}\right) \div 3600\left(\frac{sec}{hr}\right) = 2.79E+06\frac{cc}{sec}$$

Equation 7.1.2.1

Since the flow rate is set, the concentration will determine the limit. Equation 7.1.1.1 solves for the limiting concentration of 5.00E-02 $\mu Ci/cc$ as shown below.

$$\frac{Limiting\ Release\left(\frac{\mu Ci}{sec}\right)}{Release\ Rate\left(\frac{cc}{sec}\right)} = Limiting\ Concentration\left(\frac{\mu Ci}{cc}\right)$$

$$\frac{1.40 * 10^5\left(\frac{\mu Ci}{sec}\right)}{2.79 * 10^6\left(\frac{cc}{sec}\right)} = 5.00E - 02\left(\frac{\mu Ci}{cc}\right)$$

Equation 7.1.2.2

7.2 Alert, Site Area and General Emergencies – RA1, RS1, RG1

7.2.1 Unit Vent Monitor

Input

The Alert EAL is set to 10 mrem TEDE and 50 mrem Thyroid CDE per Reference 5.10. The emergency offsite dose calculation software STAMPEDE was used to calculate the release which corresponds to this dose. A release concentration correlating to the EAL threshold value was calculated by varying the input. The following assumptions and inputs were used for the calculation as described in Sections 4.0 and 6.0.

- Release begins at reactor trip
- Release lasts for one hour
- Gap inventory source term
- Default STAMPEDE input values
  - Windspeed = 13.2 mph
  - Stability class D

Results

Given a monitored unit vent release of 2.50E+06 μCi/sec, the Thyroid CDE is 51 mrem/hr at the closest portion of the site boundary and the EAL Initiating Condition is exceeded.

Threshold values for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert. Since the correlation between release concentration and dose is linear, threshold values for the steam line monitors are 2.50E+07 and 2.50E+08 μCi/sec for the SAE and GE respectively. Both are also limited by Thyroid CDE. Additional STAMPEDE iterations were performed to confirm this and are attached.

The input and output files can be found at the end of this document in Attachment 3.

7.2.2 Main Steam Line Monitor

Input

A release concentration correlating to the EAL threshold value was calculated by varying the input. The following assumptions and inputs were used for this calculation as described in Sections 4.0 and 6.0.

- Release begins at reactor trip
- Release lasts for one hour
- Noble gas + iodine with 0.2% iodine source term
- Default STAMPEDE input values
  - Windspeed = 13.2 mph
  - Stability class D

Results

Given a monitored main steam line release of 4.5 μCi/cc, the Thyroid CDE is 50 mrem/hr and the EAL Initiating Condition is exceeded.

The input and output files can be found at the end of this document in Attachment 3.

7.3 Threshold values for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert. Since the correlation between release concentration and dose is linear, threshold values for the steam line monitors are 45 and 450 μCi/cc for the SAE and GE respectively. Both are also limited by Thyroid CDE. Additional STAMPEDE iterations were performed to confirm this and are attached.

Attachment 1 – Hand Calculations

1.0 OBJECTIVE/SCOPE

Each release calculated using STAMPEDE in the main document is calculated by hand in this attachment and the results compared to STAMPEDE.

2.0 SUMMARY OF RESULTS

Table 2.1 is displayed again below showing the results from all the calculations. The minor difference is due to STAMPEDE using decay factors over a one hour period after shutdown. This also accounts for the change in the limiting dose being TEDE in the hand calculations and Thyroid CDE in the STAMPEDE calculations. The accuracy of the hand calculation is considered sufficient and recommended for use in Emergency Action Levels.

Table 2.1 Results

Emergency Action Level		RT-8010b, Unit Vent (µCi/sec)	RT-8046 through 8049, Main Steam Line (µCi/cc)
RU1	**Unusual Event**		
	Hand Calculation	1.40E+05	5.00E-02
	STAMPEDE	N/A	N/A
RA1	**Alert**		
	Hand Calculation	1.57E+06	3.90E+00
	STAMPEDE	2.50E+06	4.50E+00
RS1	**Site Area Emergency**		
	Hand Calculation	1.57E+07	3.90E+01
	STAMPEDE	2.50E+07	4.50E+01
RG1	**General Emergency**		
	Hand Calculation	1.57E+08	3.90E+02
	STAMPEDE	2.50E+08	4.50E+02

3.0 METHOD OF ANALYSIS

Using the limiting dose at the site boundary, the release is back calculated using atmospheric dispersion models. The X/Q value used is calculated from Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power* Plants. Rather than using the most conservative meteorology, average meteorological conditions are used as inputs to most closely agree with STP emergency dose assessment methodology per the ODCM and STAMPEDE. Assumed nuclide inventories are taken from Reference 5.4. The dose conversion factors are taken from Reference 5.2. A release concentration is used to find an initial projected dose at the Site Boundary. Using the projected dose at the Site Boundary, the release concentration is scaled to find the limiting dose for each EAL.

4.0 INPUTS

- The Unit Vent flow rate is taken from the Offsite Dose Calculation Manual; Revision 17, March 2011 and is 9.44E+07 cc/sec.
- The main steam line pressure and PORV choke flow rate were taken from Reference 5.5 and are 1285 psig and 1.05E+06 lbm/hr respectively.
- The specific volume of saturated steam at this pressure is taken from the NIST steam tables and is 0.338 ft^3/lbm.
- The release concentration is varied to find the release concentration which correlates to each emergency action level dose. Emergency action level doses are taken from NEI 99-01 Revision 6 for initiating conditions AA1, AS1 and AG1. EAL 1 is the EAL of interest in each initiating condition. The limiting doses are listed in Table 4.1. NEI 990-01 Revision 6 states that these values are based on fractions of the Environmental Protection Agencies Protective Action Guidelines (EPA PAGs) and the General Emergency represents the protective action values recommended by the EPA.

Table 4.1 EAL Thresholds

	Alert	Site Area	General
TEDE	10 mrem	100 mrem	1000 mrem
Thyroid CDE	50 mrem	500 mrem	5000 mrem

- A release lasting one hour is selected per NEI 99-01 Revision 6 developer notes.
- Atmospheric dispersion factors are calculated per Regulatory Guide 1.145 (Reference 5.1). The reactor building dimensions used as inputs for this calculation are taken from Reference 5.13.
- Nuclide inventories are taken from TGX/THX 3-1, (Reference 5.4) which is the source document for the nuclide inventories used in STAMPEDE. The release inventories are a gap release and noble gases plus 0.2% iodine which are listed below. Each nuclide inventory was normalized to one so it could be scaled to various release activities.

Table 4.2 Gap Inventory

Nuclide	Activity (µCi/cc)	Normalized	Nuclide	Activity (µCi/cc)	Normalized
I-131	1.10E+05	1.12E-03	Xe-135	5.50E+06	5.62E-02
I-132	1.50E+05	1.53E-03	Xe-137	1.90E+07	1.94E-01
I-133	2.20E+05	2.25E-03	Xe-138	1.80E+07	1.84E-01
I-134	2.40E+05	2.45E-03	Cs-134	3.70E+01	3.78E-07
I-135	2.00E+05	2.05E-03	Cs-137	2.90E+01	2.97E-07
Kr-83m	1.30E+06	1.33E-02	Te132	4.80E+00	4.91E-08
Kr-85m	2.90E+06	2.97E-02	Mo99	1.22E+01	1.25E-07
Kr-85	3.70E+05	3.78E-03	Ru103	8.80E-03	9.00E-11
Kr-87	5.50E+06	5.62E-02	Ru106	2.90E-03	2.97E-11
Kr-88	7.80E+06	7.98E-02	Zr95	1.10E-02	1.12E-10
Kr-89	9.50E+06	9.72E-02	La140	1.90E-02	1.94E-10
Xe-131m	1.10E+05	1.12E-03	Ce144	7.40E-03	7.57E-11
Xe-133m	6.80E+05	6.95E-03	Ce-141	1.00E-02	1.02E-10
Xe-133	2.20E+07	2.25E-01	Sr89	6.40E-02	6.55E-10
Xe-135m	4.20E+06	4.30E-02	Sr90	3.20E-03	3.27E-11

Table 4.3 Noble Gases+0.2% Iodine Inventory

Nuclide	Inventory	Normalized
I-131	6.10E-02	2.26E-04
I-132	8.61E-02	3.19E-04
I-133	1.00E-01	3.72E-04
I-134	1.86E-02	6.92E-05
I-135	2.73E-01	1.01E-03
Xe-131m	2.80E+00	1.04E-02
Xe-133	2.40E+02	8.90E-01
Xe-133m	4.20E+00	1.56E-02
Xe-135	7.60E+00	2.82E-02
Xe-135m	4.00E-01	1.48E-03
Xe-137	1.60E-01	5.93E-04
Xe-138	5.80E-01	2.15E-03
Kr-83m	3.70E-01	1.37E-03
Kr-85	7.60E+00	2.82E-02
Kr-85m	1.50E+00	5.56E-03
Kr-87	9.80E-01	3.63E-03
Kr-88	2.80E+00	1.04E-02
Kr-89	8.40E-02	3.12E-04

- The dose conversion factors taken from EPA 400R92001 (Reference 5.2) are listed in Tables 4.4 and 4.5 below.

Table 4.4 TEDE Dose Conversion Factors

Nuclide	Dose Conversion Factor (rem per uCi*hr/cc)	Nuclide	Dose Conversion Factor (rem per uCi*hr/cc)
I-131	5.30E+04	**Xe-135**	1.40E+02
I-132	4.90E+03	**Xe-137**	1.10E+02
I-133	1.50E+04	**Xe-138**	7.20E+02
I-134	3.10E+03	**Cs-134**	6.30E+04
I-135	8.10E+03	**Cs-137**	4.10E+04
Kr-83m		**Te132**	1.20E+04
Kr-85m	9.30E+01	**Mo99**	5.20E+03
Kr-85	1.30E+00	**Ru103**	1.30E+04
Kr-87	5.10E+02	**Ru106**	5.70E+05
Kr-88	1.30E+03	**Zr95**	3.20E+04
Kr-89	1.20E+03	**La140**	1.10E+04
Xe-131m	4.9	**Ce144**	4.50E+05
Xe-133m	1.70E+01	**Ce-141**	1.10E+04
Xe-133	2.00E+01	**Sr89**	5.00E+04
Xe-135m	2.50E+02	**Sr90**	1.60E+06

Table 4.5 Thyroid CDE Dose Conversion Factors

Nuclide	Thyroid CDE DCF (rem per uCi*hr/cc)
I-131	1.30E+06
I-132	7.70E+03
I-133	2.20E+05
I-134	1.30E+03
I-135	3.80E+04

- The unit vent noble gas monitor energy efficiency by nuclide is taken from Offsite Dose Calculation Manual (Reference 5.3). The values are relative to Xe-133 efficiency since the monitor is calibrated to Xe-133. Table 4.6 displays the energy efficiency by nuclide relative to Xe-133.

Table 4.6 Energy Efficiency Relative to Xe-133

Nuclide	Efficiency Relative to Xe-133 $(uCi/cc)_{equivalent}$
Kr-83m	*
Kr-85m	1.9
Kr-85	2.4
Kr-87	2.8
Kr-88	2.3
Kr-89	2.8
Xe-131m	0.015
Xe-133m	0.14
Xe-133	1
Xe-135m	0.042
Xe-135	2.5
Xe-137	2.8
Xe-138	2.8

*There is no relative efficiency available for Kr-83m. Assumption 6.4 further justifies the omission.

Table 4.7 Nuclide Half Lives

Nuclide	Half Life (hr)	Nuclide	Half Life (hr)
I-131	1.93E+02	Xe-135	9.08E+00
I-132	2.38E+00	Xe-137	6.38E-02
I-133	2.03E+01	Xe-138	2.36E-01
I-134	8.77E-01	Cs-134	1.80E+04
I-135	6.61E+00	Cs-137	2.60E+05
Kr-83m	1.83E+00	Te132	7.79E+01
Kr-85m	4.48E+00	Mo99	6.62E+01
Kr-85	9.40E+04	Ru103	9.44E+02
Kr-87	1.27E+00	Ru106	8.84E+03
Kr-88	2.84E+00	Zr95	1.55E+03
Kr-89	5.10E-02	La140	4.03E+01
Xe-131m	2.83E+02	Ce144	6.82E+03
Xe-133m	5.42E+01	Ce-141	7.77E+02
Xe-133	1.27E+02	Sr89	1.21E+03
Xe-135m	2.60E-01	Sr90	2.50E+05

- The half-lives are taken from Reference 5.15 which lists the input data used by STAMPEDE.

5.0 REFERENCES

5.1 Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, Revision 1, November 1982.
5.2 EPA 400R92001, Manual of Protective Action Guides and Protective actions for Nuclear Incidents, Revision 1, May 1992.
5.3 Offsite Dose Calculation Manual, Revision 17, March 2011.
5.4 TGX/THX 3-1, Revision 5, Westinghouse Radiation Analysis Manual.
5.5 MC05591, Main Steam PORV Capacity Verification, Revision 1.
5.6 NIST Steam Tables, 2011.
5.7 0ERP01-ZV-IN01, Emergency Classification, Revision 10.
5.8 0ERP01-ZV-TP01, Offsite Dose Calculations, Revision 21.
5.9 STP Calculation NC-9012, Process and Effluent Radiation Monitor Set Points, Revision 7
5.10 STP Calculation NC-9011, CRMS Rad Monitor Setpoints, Revision 2.
5.11 STAMPEDE Computer Program, Revision 7.0.3.3.
5.12 STAMPEDE User's Manual
5.13 STP Drawing 6C189N5007, General Arrangement Reactor Containment Building, Revision 6
5.14 NEI 99-01, Revision 6, Development of Emergency Action Levels for Non-Passive Reactors
5.15 ITWMS Call Number 1000010987 Design Document, Revision 0

6.0 ASSUMPTIONS

6.1 Release lasts for one hour

Per NEI 99-01 (Reference 5.14), IC AA1, AS1, AG1 developer notes, the release should be assumed to last one hour.

For this to be true for the main steam line, it is assumed that the PORV is open for one hour. To calculate the most limiting case, it is assumed that the maximum flow possible is being released from the PORV.

6.2 Nuclide mix

Per 0ERP01-ZV-TP01, Offsite Dose Calculations (Reference 5.8) any unit vent release with increased RCS activity and no core melt should be calculated using a gap inventory. It is conservative to assume an increased RCS activity and not within the intended scope of the relevant initiating conditions to assume core melt. Therefore, a gap inventory is used for all unit vent releases.

Per 0ERP01-ZV-TP01, Offsite Dose Calculations (Reference 5.8) for a main steam line release following a steam generator tube rupture it is appropriate to use an inventory of 100 percent noble gases plus 0.2 percent iodine. Since a steam generator tube rupture releasing through the PORVs is the only steam generator tube rupture scenario which would create offsite doses large enough to meet or exceed the EALs, this assumption is made.

6.3 Atmospheric Dispersion

NEI 99-01 (Reference 5.14) developer notes for initiating conditions AA1, AS1 and AG1 suggest using the ODCM or the site's emergency dose assessment methodology. Per 0ERP01-ZV-TP01, Offsite Dose Calculations (Reference 5.8), when actual meteorology is not available, the default STAMPEDE values should be used. The default STAMPEDE values assume a stability class D for atmospheric dispersion and a windspeed of 13.2 mph. These values were used as inputs for the atmospheric dispersion calculation.

It is clear that STAMPEDE uses the same method for calculating atmospheric dispersion factor (X/Q) outlined in section 7.1.1 of this Attachment. However, STAMPEDE does not follow the same logic in selecting the appropriate result from the three calculations. The STAMPEDE value printed in the results found in attachment 3 is consistent with the largest of the three hand calculated X/Q values. This suggests that STAMPEDE simply selects the largest of the three X/Q values resulting in a much more conservative estimate. This calculation will deviate from the recommendations of Regulatory Guide 1.145 and conform to the methodology STAMPEDE uses.

The close proximity of all release points allows for a single atmospheric dispersion coefficient to be used. This assumption is also made by STAMPEDE.

6.4 Exposure Pathways

The dose conversion factors used in table 4.4 and 4.5 represent a summation of dose conversion factors for external plume exposure, inhalation from the plume, and external exposure from deposition. Because the dose estimations are used for implementing early phase protective actions, conversion factors using limited pathways are appropriate.

The EPA does not provide a dose conversion factor for Kr-83m. Because the PAGs are based on EPA dose calculations, it is appropriate to only use the nuclides for which dose conversion factors are provided. Additionally, Kr-83m represents only 1.33% of the nuclide inventory activity and its exclusion would not significantly affect the final dose.

6.5 The release initiates one hour after reactor shutdown. While a release initiating at reactor shutdown is likely, significant decay of short lived nuclides occurs during the migration time. A release at reactor shutdown would have a significantly higher activity at the monitor location than at the reception site. It is important for the threshold to not be calculated at shutdown as this would create a very high threshold which would not be appropriate for releases which occur shortly after shutdown. One hour after reactor shutdown is sufficient time to decay short lived nuclides and create a conservative threshold.

Decay is incorporated for one hour from reactor shutdown as well as migration time. Half-lives are taken from Reference 5.15. Migration time is assumed to be the reciprocal of the wind speed.

7.0 HAND CALCULATIONS

7.1 Unit Vent Monitor

7.1.1 X/Q

The atmospheric dispersion factor, X/Q, determines the change in concentration between the unit vent discharge and the dose reception site. This value is based on meteorological conditions and will vary with wind speed and stability class. The ODCM uses the highest annual average X/Q value at the site boundary which is 5.3E-06 sec/m^3. However, for an accident related release STAMPEDE is used rather than the ODCM. STAMPEDE uses real time, user entered, or default meteorological conditions to calculate the X/Q for a specific accident. Default values will be used as inputs into the Regulatory Guide 1.145 method for calculating X/Q as described below. Default values are identified in section 6.0, Atmospheric Dispersion.

For a neutral atmospheric stability class, which is the default in STAMPEDE, X/Q values can be determined through the following set of equations.

$$\frac{X}{Q} = \frac{1}{\bar{U}_{10}\left(\pi\sigma_y\sigma_z + \frac{A}{2}\right)}$$

Equation 7.1.1.1

$$\frac{X}{Q} = \frac{1}{\bar{U}_{10}(3\pi\sigma_y\sigma_z)}$$

Equation 7.1.1.2

$$\frac{X}{Q} = \frac{1}{\bar{U}_{10}\pi\Sigma_y\sigma_z}$$

Equation 7.1.1.3

Where

X/Q = relative concentration (sec/m^3)

π = 3.14159

$\bar{U}_{10}$ = windspeed at 10 meters above plant grade (m/s)

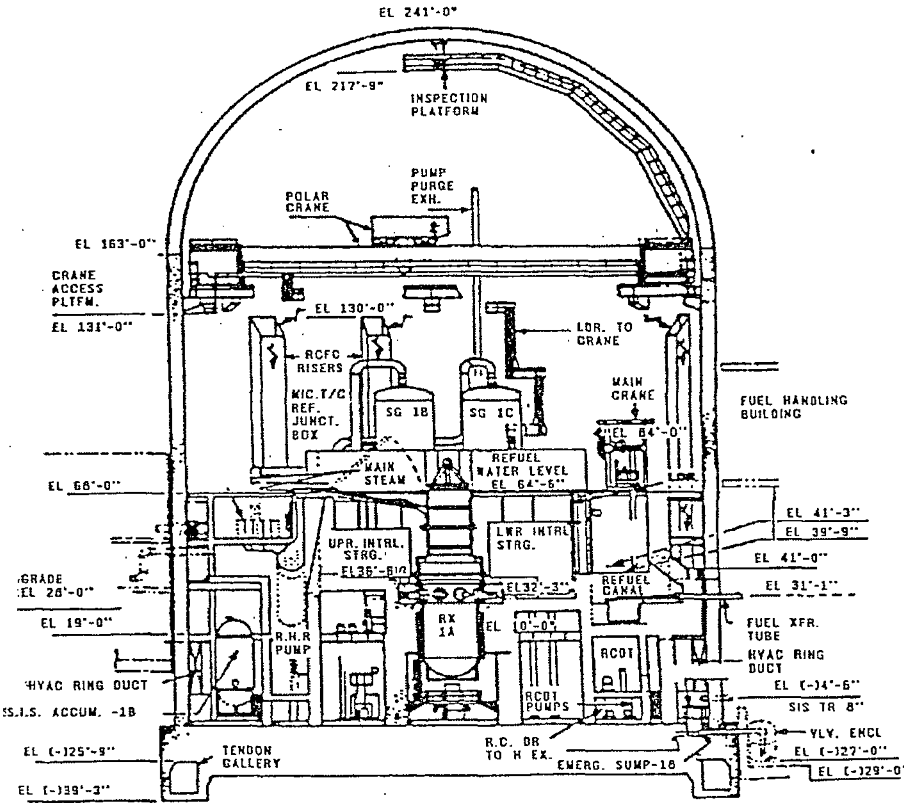
σ_y = lateral plume spread (m), a function of atmospheric stability and distance, determined from Regulatory Guide 1.145 Figure 1

σ_z = vertical plume spread (m), a function of atmospheric stability and distance, determined from Regulatory Guide 1.145 Figure 2

Σ_y = $(M-1)\sigma_{y800m} + \sigma_y$ = lateral plume spread with meander and building wake effects (m), a function of atmospheric stability, windspeed $\bar{U}_{10}$, and distance; M is determined from Regulatory Guide 1.145 Figure 3

A = the smallest vertical-plane cross-sectional area of the reactor building (m^2), taken from Reference 5.13 and shown below

Figure 7.1.1.1: Reactor Building Dimensions



Assuming the reactor building cross section to be a perfect rectangle and half sphere, the variables are defined as follows;

$\bar{U}_{10}$ = 13.2 mph = 5.9 m/s

σ_y = 1200 m

σ_z = 4.2 m

$\Sigma_y = (M-1)\sigma_{y800m} + \sigma_y$; M=1 $\rightarrow \sigma_y$ = 1200 m

A $= (135' * 158') + \left(\frac{\pi * 79^2}{2}\right) = 31128.37$

The three equations become;

$$\frac{X}{Q} = \frac{1}{5.9\left(\pi 1200 * 4.2 * \frac{31128.37}{2}\right)} = 5.398 * 10^{-6}$$

$$\frac{X}{Q} = \frac{1}{5.9(3\pi * 1200 * 4.2)} = 3.568 * 10^{-6}$$

$$\frac{X}{Q} = \frac{1}{5.9 * \pi * [(1-1)\sigma_{y800m} + 1200] * 4.2} = 1.07 * 10^{-5}$$

To select the appropriate X/Q value, the first two X/Q values should be compared and the higher value selected. This value is then compared with the third X/Q value and the lower of those two is the appropriate X/Q value. The appropriate X/Q is 5.39E-06 sec/m^3 for default meteorological conditions by the methodology recommended in Regulatory Guide 1.145.

This calculated value is very similar to the ODCM highest average value of 5.3E-06 sec/m^3 which was not selected for use. Additionally, the value shown in the STAMPEDE output file at one mile is 1.032E-05 sec/m^3. This suggests that STAMPEDE uses the same methodology and simply selects the largest atmospheric dispersion value to remain conservative. This methodology will be replicated and 1.07E-05 will be used as the X/Q.

7.1.2 Nuclide Inventory

As previously stated, a gap inventory is appropriate for this problem. The gap inventory is taken from TGX/THX 3-1 (Reference 5.4) which is used as the source term for STAMPEDE inventories. The concentrations were then normalized so they could be scaled to the varying emergency classifications. The values for the normalized inventory can be found in Table 4.2.

7.1.3 Dose Conversion Factors

As stated in NEI99-01 (Reference 5.14) developer notes, the purpose of dose projections is to check if the Environmental Protection Agencies Protective Action Guidelines (EPA PAGs) have been exceeded. The dose conversion factors provided by the EPA in EPA 400R92001 are used. These dose conversion factors account for external plume exposure, inhalation from the plume, and external exposure from deposition and are listed Tables 4.4 and 4.5, and taken from tables 5-1, 5-2 in EPA 400R92001 (Reference 5.2).

The EPA does not provide a dose conversion factor for Kr-83m. This nuclide contributes 1.33% of the inventory activity. The lack of this nuclide's contribution to the final dose will not significantly affect the outcome.

7.1.4 Decay Time

One hour of decay is incorporated to the monitor response due to the release initiating one hour after reactor shutdown. Decay is also incorporated for the duration of the migration time. The total decay time is one hour plus the reciprocal of wind speed, or 1.07575 hours.

7.1.5 Dose Calculations

The dose rate at the site boundary is calculated using Equation 7.1.5.1.

$$\dot{D} = \frac{X}{Q} F \sum_{i}^{n} C_i * 0.5^{\frac{1.07575}{T_{1/2_i}}} * DCF_i$$

Equation 7.1.5.1

Where

$\dot{D}$ = dose rate per hour at the site boundary

$\frac{X}{Q}$ = atmospheric dispersion coefficient as calculated in section 7.1.1

F = unit vent flow rate

C_i= concentration of nuclide i at the time of shutdown

1.07575 = the total decay time of interest from section 7.1.4

$T_{1/2i}$ = the half-life of nuclide i

DCF_i = the dose conversion factor for nuclide i listed in tables 4.4 and 4.5

The total concentration of the nuclides is varied to find the dose rate of interest. Beginning with an arbitrary release concentration of 1 µCi/cc, the dose rate is calculated. Since the dose is linearly correlated to concentration, the release concentration may be scaled to find the dose rate of interest.

The Alert EAL is 10 mrem TEDE or 50 mrem Thyroid CDE. Using the above method to calculate TEDE with the appropriate conversion factors, a limiting release rate of 2.33E+06 µCi/sec from the unit vent results in 5.7 mrem TEDE. Using the calculated release rate to find Thyroid CDE with the appropriate conversion factors, the same release results in a 50 mrem Thyroid CDE at the site boundary. Thus, 2.33E+06 µCi/sec is the limiting release rate based on the 50 mrem Thyroid CDE EAL initiating condition.

The limiting release rate threshold values for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert release rate threshold value.

These calculations can be found in Attachment 2.

7.1.6 Monitor Response

The unit vent noble gas monitor is calibrated to Xe-133. Monitor efficiencies relative to Xe-133 by nuclide are listed in ODCM Table B3-2. To find the monitor reading associated with each limiting release, the noble gas concentrations must be multiplied by the monitor response and summed. Table 4.6 shows the indicated response of the unit vent noble gas monitor by nuclide and Equation 7.1.5.1 shows how the monitor response was calculated.

$$Monitor\ Response = \sum_{i}^{n} C_i * Re_i$$

Equation 7.1.5.1

Where

C_i = concentration of nuclide i (μCi/cc)

Re_i = monitor response to nuclide i $(\mu Ci/cc)_{Xe\text{-}133\ equivalent}$

In the case of an Alert, the 2.33E+06 μCi/sec release rate will read as 1.57E+06 μCi/sec on the monitor. Kr-83m does not have an indicated monitor response coefficient. Because Kr-83m is only 1.34% of the noble gases and does not contribute to the dose calculation, its exclusion is acceptable.

This again is a linear correlation and the SAE and GE scale by factors of 10 and 100 respectively.

These calculations can be found in Attachment 2.

7.2 Main Steam Line Monitors

7.2.1 X/Q

Since the atmospheric dispersion is independent of nuclide inventory or release rate and the close proximity of the releases, the X/Q value will be the same for a main steam line release as it is for a unit vent release. This assumption is also taken by STAMPEDE and outline in Assumption 6.3.

7.2.2 Nuclide Inventory

Per 0ERP01-ZV-TP01, if the release path is the main steam line with a steam generator tube rupture, the nuclide inventory should be 100% noble gas and 0.2% of the iodine from the reactor coolant.

The secondary steam concentration for noble gases and iodine after a steam generator tube rupture are taken from TGX/THX 3-1 (Reference 5.4). Values for the reactor coolant inventory are listed in table 4.3. All of the noble gases are used and the iodine concentration from the coolant inventory is scaled to total 0.2% of iodine in the total coolant inventory. These inventories are then normalized to one. These values are listed in Table 4.3.

7.2.3 Dose Conversion Factors

The dose conversion factors used are found in Tables 4.4 and 4.5, taken from tables 5-1, 5-2 in EPA 400R92001.

7.2.4 Decay Time

One hour of decay is incorporated to the monitor response due to the release initiating one hour after reactor shutdown. Decay is also incorporated for the duration of the migration time. The total decay time is one hour plus the reciprocal of wind speed, or 1.07575 hours.

7.2.5 Dose Calculations

Equation 7.1.5.1 applies to the release from the main steam lines. The main steam line flow rate is used instead of the unit vent flow rate for the value F. The main steam line flow rate was calculated in Equation 7.1.2.2 of the STAMPEDE CALCULATIONS section of this document as 2.79E+06 cc/sec.

The Alert EAL threshold is 10 mrem TEDE or 50 mrem Thyroid CDE at the site boundary (Table 4.2). Using the method in Equation 7.1.5.1 to calculate TEDE with the appropriate conversion factors, a concentration at time of shutdown of 4.10 μCi/cc would result in 6.89 mrem TEDE at the site boundary if the steam line PORV was open for an hour. Using the same steam line concentration to calculate Thyroid CDE results in 50 mrem Thyroid CDE at the site boundary.

The steam line concentrations at the time of shutdown for the Site Area Emergency and General Emergencies are multiples of 10 and 100 of the Alert. Since the correlation between release concentration and dose is linear, values for the steam line concentration at time of shutdown are 41.0 and 410 μCi/cc for the SAE and GE respectively. Both are also limited by Thyroid CDE.

These calculations can be found in Attachment 2.

7.2.6 Monitor Response

Because the main steam line monitor is adjacent to the main steam line, significant shielding takes place between the source and monitor. STP calculation NC-9011 Revision 2 calculates a conversion factor for the main steam lines for a noble gas inventory which is incorporated into the monitor readout. No monitor response needs to be calculated.

The concentration of the main steam line one hour after shutdown given a concentration of 4.10 μCi/cc at the time of shutdown is 3.90 μCi/cc. This calculation is also found in Attachment 2. Additionally, the monitor readings for the SAE and GE one hour after shutdown are 39.0 and 390 μCi/cc respectively. These values are the thresholds for the main steam line monitor.

Table A2-1: Unusual Event Emergency Calculations

Unit Vent Limiting Concentration	Flow Rate	Release Rate
(uCi/cc)	(cc/sec)	(uCi/sec)
1.48E-03	9.44E+07	**1.40E+05**
MSL Limiting Release Rate	**Flow Rate**	**Limiting Concentration**
(uCi/sec)	(cc/sec)	(uCi/cc)
1.40E+05	2.79E+06	**5.00E-02**

Table A2-2: Input Values for Calculations

X/Q	duration	Release Rate	Release Constant	Unit Conversion for Release Constant	Total Release Variable	Decay Time
(s/m^3)	(s)	(cc/sec)	(s*cc/m^3)	(hr*m^3/s*cc)	(uCi/cc)	(hr)
5.40E-06	3600	9.44E+07	1.83E+06	5.10E-04	1.79E-02	1.07575

Table A2-3: Calculations for Boundary Concentrations and TEDE dose due to Unit Vent Release

Nuclide	Inventory	Normalized	Varied concentration	Release Constant	Concentration @ Boundary	Half Life	Decayed Concentration	Dose Conversion Factor	Dose Contribution
	µCi/cc		(uCi/cc)	(hr*cc/cc)	(uCi*hr/cc)	(hr)	(µCi*hr/cc)	rem per uCi*hr/cc	(rem)
I-131	1.10E+05	1.12E-03	2.76E-05	1.01E-03	2.79E-08	1.93E+02	2.78E-08	5.30E+04	1.47E-03
I-132	1.50E+05	1.53E-03	3.77E-05	1.01E-03	3.81E-08	2.38E+00	2.79E-08	4.90E+03	1.37E-04
I-133	2.20E+05	2.25E-03	5.55E-05	1.01E-03	5.61E-08	2.03E+01	5.40E-08	1.50E+04	8.11E-04
I-134	2.40E+05	2.45E-03	6.04E-05	1.01E-03	6.11E-08	8.77E-01	2.61E-08	3.10E+03	8.09E-05
I-135	2.00E+05	2.05E-03	5.06E-05	1.01E-03	5.11E-08	6.61E+00	4.56E-08	8.10E+03	3.70E-04
Kr-83m	1.30E+06	1.33E-02	3.28E-04	1.01E-03	3.31E-07	1.83E+00	2.21E-07		0.00E+00
Kr-85m	2.90E+06	2.97E-02	7.33E-04	1.01E-03	7.40E-07	4.48E+00	6.27E-07	9.30E+01	5.83E-05
Kr-85	3.70E+05	3.78E-03	9.33E-05	1.01E-03	9.42E-08	9.40E+04	9.42E-08	1.30E+00	1.22E-07
Kr-87	5.50E+06	5.62E-02	1.39E-03	1.01E-03	1.40E-06	1.27E+00	7.79E-07	5.10E+02	3.97E-04
Kr-88	7.80E+06	7.98E-02	1.97E-03	1.01E-03	1.99E-06	2.84E+00	1.53E-06	1.30E+03	1.99E-03
Kr-89	9.50E+06	9.72E-02	2.40E-03	1.01E-03	2.42E-06	5.10E-02	1.08E-12	1.20E+03	1.30E-09
Xe-131m	1.10E+05	1.12E-03	2.76E-05	1.01E-03	2.79E-08	2.83E+02	2.78E-08	2.50E+02	1.36E-07
Xe-133m	6.80E+05	6.95E-03	1.71E-04	1.01E-03	1.73E-07	5.42E+01	1.71E-07	1.40E+02	2.90E-06
Xe-133	2.20E+07	2.25E-01	5.55E-03	1.01E-03	5.61E-06	1.27E+02	5.57E-06	1.10E+02	1.11E-04
Xe-135m	4.20E+06	4.30E-02	1.06E-03	1.01E-03	1.07E-06	2.60E-01	6.09E-08	7.20E+02	1.52E-05
Xe-135	5.50E+06	5.62E-02	1.39E-03	1.01E-03	1.40E-06	9.08E+00	1.29E-06	5.30E+04	1.81E-04
Xe-137	1.90E+07	1.94E-01	4.79E-03	1.01E-03	4.83E-06	6.38E-02	4.06E-11	4.90E+03	4.47E-09
Xe-138	1.80E+07	1.84E-01	4.54E-03	1.01E-03	4.59E-06	2.36E-01	1.95E-07	1.50E+04	1.40E-04

Nuclide	Inventory	Normalized	Varied concentration	Release Constant	Concentration @ Boundary	Half Life	Decayed Concentration	Dose Conversion Factor	Dose Contribution
	µCi/cc		(µCi/cc)	(hr*cc/cc)	(uCi*hr/cc)	(hr)	(µCi*hr/cc)	rem per uCi*hr/cc	(rem)
Cs-134	3.70E+01	3.78E-07	9.33E-09	1.01E-03	9.42E-12	1.80E+04	9.42E-12	6.30E+04	5.93E-07
Cs-137	2.90E+01	2.97E-07	7.33E-09	1.01E-03	7.40E-12	2.60E+05	7.40E-12	4.10E+04	3.03E-07
Te132	4.80E+00	4.91E-08	1.21E-09	1.01E-03	1.22E-12	7.79E+01	1.21E-12	1.20E+04	1.45E-08
Mo99	1.22E+01	1.25E-07	3.08E-09	1.01E-03	3.11E-12	6.62E+01	3.08E-12	5.20E+03	1.60E-08
Ru103	8.80E-03	9.00E-11	2.22E-12	1.01E-03	2.24E-15	9.44E+02	2.24E-15	1.30E+04	2.91E-11
Ru106	2.90E-03	2.97E-11	7.33E-13	1.01E-03	7.40E-16	8.84E+03	7.40E-16	5.70E+05	4.22E-10
Zr95	1.10E-02	1.12E-10	2.76E-12	1.01E-03	2.79E-15	1.55E+03	2.79E-15	3.20E+04	8.93E-11
La140	1.90E-02	1.94E-10	4.79E-12	1.01E-03	4.83E-15	4.03E+01	4.75E-15	1.10E+04	5.22E-11
Ce144	7.40E-03	7.57E-11	1.87E-12	1.01E-03	1.89E-15	6.82E+03	1.89E-15	4.50E+05	8.49E-10
Ce-141	1.00E-02	1.02E-10	2.52E-12	1.01E-03	2.54E-15	7.77E+02	2.54E-15	1.10E+04	2.79E-11
Sr89	6.40E-02	6.55E-10	1.62E-11	1.01E-03	1.63E-14	1.21E+03	1.63E-14	5.00E+04	8.16E-10
Sr90	3.20E-03	3.27E-11	8.07E-13	1.01E-03	8.15E-16	2.50E+05	8.15E-16	1.60E+06	1.30E-09
								Total TEDE Dose	**5.77E-03**

Table A2-4: Thyroid Dose Calculation for Unit Vent Release

Nuclide	Decayed Concentration	Thyroid DCF	Thyroid Dose
	(μCi*hr/cc)	rem per μCi*hr/cc	(rem)
I-131	2.78E-08	1.30E+06	3.61E-02
I-132	2.79E-08	7.70E+03	2.15E-04
I-133	5.40E-08	2.20E+05	1.19E-02
I-134	2.61E-08	1.30E+03	3.39E-05
I-135	4.56E-08	3.80E+04	1.73E-03
		Total Thyroid Dose	5.00E-02

Table A2-5: Unit Vent Monitor Response to Nuclide Inventory

Nuclide	Concentration	Half Life	Concentration After 1 hr	Response Coeff	Response	
	(μCi/cc)	Hours	(μCi/cc)	(μCi/cc)	(μCi/cc)	
Kr-83m	3.28E-04	1.83E+00	2.25E-04		0.00E+00	
Kr-85m	7.33E-04	4.48E+00	6.28E-04	1.9	1.19E-03	
Kr-85	9.33E-05	9.40E+04	9.33E-05	2.4	2.24E-04	
Kr-87	1.39E-03	1.27E+00	8.03E-04	2.8	2.25E-03	
Kr-88	1.97E-03	2.84E+00	1.54E-03	2.3	3.55E-03	
Kr-89	2.40E-03	5.10E-02	3.00E-09	2.8	8.40E-09	
Xe-131m	2.76E-05	2.83E+02	2.76E-05	0.015	4.13E-07	
Xe-133m	1.71E-04	5.42E+01	1.69E-04	0.14	2.37E-05	
Xe-133	5.55E-03	1.27E+02	5.52E-03	1	5.52E-03	
Xe-135m	1.06E-03	2.60E-01	7.38E-05	0.042	3.10E-06	
Xe-135	1.39E-03	9.08E+00	1.28E-03	2.5	3.21E-03	
Xe-137	4.79E-03	6.38E-02	9.15E-08	2.8	2.56E-07	
Xe-138	4.54E-03	2.36E-01	2.41E-04	2.8	6.74E-04	
				Monitor Reading:	1.66E-02	1.57E+06
					(uCi/cc)	(uCi/sec)

Table A2-6: Input for Main Steam Line Release Calculation

Choke Flow	Limiting Release	Steam Specific Volume	Volumetric Release	Release Rate unit conversion	Variable concentration	Decay Time
(lbm/hr)	(uCi/sec)	(ft^3/lbm)	(ft^3/hr)	(cc/sec)	(µCi/cc)	(hr)
1.05E+06	3.37E+06	0.338	3.55E+05	2.79E+06	4.05	1.07575

Table A2-7: Calculations for Boundary Concentrations and TEDE dose due to Main Steam Line Release

Nuclide	Steam Inventory	Normalized	Varied Concentration	Release Constant*	Concentration @ Boundary	Half life	Decayed Concentration	DCF	Dose Contribution
			(µCi/cc)	(hr*cc/cc)	(µCi*hr/cc)	(hr)	(µCi*hr/cc)	(rem per µCi*hr/cc)	(rem)
I-131	6.10E-02	2.26E-04	9.27E-04	2.9853E-05	2.77E-08	1.93E+02	2.76E-08	5.30E+04	1.46E-03
I-132	8.61E-02	3.19E-04	1.31E-03	2.9853E-05	3.90E-08	2.38E+00	2.85E-08	4.90E+03	1.40E-04
I-133	1.00E-01	3.72E-04	1.53E-03	2.9853E-05	4.55E-08	2.03E+01	4.39E-08	1.50E+04	6.58E-04
I-134	1.86E-02	6.92E-05	2.84E-04	2.9853E-05	8.47E-09	8.77E-01	3.62E-09	3.10E+03	1.12E-05
I-135	2.73E-01	1.01E-03	4.14E-03	2.9853E-05	1.24E-07	6.61E+00	1.10E-07	8.10E+03	8.95E-04
Xe-131m	2.80E+00	1.04E-02	4.26E-02	2.9853E-05	1.27E-06	2.83E+02	1.27E-06	4.90E+00	6.22E-06
Xe-133	2.40E+02	8.90E-01	3.65E+00	2.9853E-05	1.09E-04	5.42E+01	1.07E-04	2.00E+01	2.15E-03
Xe-133m	4.20E+00	1.56E-02	6.40E-02	2.9853E-05	1.91E-06	1.27E+02	1.90E-06	1.70E+01	3.23E-05
Xe-135	7.60E+00	2.82E-02	1.16E-01	2.9853E-05	3.45E-06	2.60E-01	1.96E-07	1.40E+02	2.75E-05
Xe-135m	4.00E-01	1.48E-03	6.07E-03	2.9853E-05	1.81E-07	9.08E+00	1.67E-07	2.50E+02	4.17E-05
Xe-137	1.60E-01	5.93E-04	2.43E-03	2.9853E-05	7.26E-08	6.38E-02	6.10E-13	1.40E+02	8.53E-11
Xe-138	5.80E-01	2.15E-03	8.82E-03	2.9853E-05	2.63E-07	2.36E-01	1.12E-08	7.20E+02	8.04E-06
Kr-83m	3.70E-01	1.37E-03	5.62E-03	2.9853E-05	1.68E-07	1.83E+00	1.12E-07		0.00E+00
Kr-85	7.60E+00	2.82E-02	1.16E-01	2.9853E-05	3.45E-06	4.48E+00	2.92E-06	1.30E+00	3.80E-06
Kr-85m	1.50E+00	5.56E-03	2.28E-02	2.9853E-05	6.81E-07	9.40E+04	6.81E-07	9.30E+01	6.33E-05
Kr-87	9.80E-01	3.63E-03	1.49E-02	2.9853E-05	4.44E-07	1.27E+00	2.47E-07	5.10E+02	1.26E-04
Kr-88	2.80E+00	1.04E-02	4.26E-02	2.9853E-05	1.27E-06	2.84E+00	9.79E-07	1.30E+03	1.27E-03
Kr-89	8.40E-02	3.12E-04	1.28E-03	2.9853E-05	3.82E-08	5.10E-02	1.71E-14	1.20E+03	2.05E-11
								Total Dose	**6.89E-03**

*Release Constant = X/Q * duration * release rate

Table A2-8: Main Steam Line Release Thyroid Dose Calculation

Nuclide	Concentration @Boundary (μCi*hr/cc)	Thyroid DCF rem per uCi*hr/cc	Thyroid Dose (rem)
I-131	2.76E-08	1.30E+06	3.58E-02
I-132	2.85E-08	7.70E+03	2.20E-04
I-133	4.39E-08	2.20E+05	9.66E-03
I-134	3.62E-09	1.30E+03	4.71E-06
I-135	1.10E-07	3.80E+04	4.20E-03
			4.99E-02

Table A2-9: Main Steam Line Reading at Release

Nuclide	Concentration	Half Life	Concentration 1 Hour after Shutdown
	(µCi/cc)	Hours	(µCi/cc)
I-131	9.27E-04	1.93E+02	9.23E-04
I-132	1.31E-03	2.38E+00	9.77E-04
I-133	1.53E-03	2.03E+01	1.47E-03
I-134	2.84E-04	8.77E-01	1.29E-04
I-135	4.14E-03	6.61E+00	3.73E-03
Xe-131m	4.26E-02	2.83E+02	4.25E-02
Xe-133	3.65E+00	5.42E+01	3.60E+00
Xe-133m	6.40E-02	1.27E+02	6.36E-02
Xe-135	1.16E-01	2.60E-01	8.04E-03
Xe-135m	6.07E-03	9.08E+00	5.62E-03
Xe-137	2.43E-03	6.38E-02	4.65E-08
Xe-138	8.82E-03	2.36E-01	4.67E-04
Kr-83m	5.62E-03	1.83E+00	3.85E-03
Kr-85	1.16E-01	4.48E+00	9.90E-02
Kr-85m	2.28E-02	9.40E+04	2.28E-02
Kr-87	1.49E-02	1.27E+00	8.62E-03
Kr-88	4.26E-02	2.84E+00	3.34E-02
Kr-89	1.28E-03	5.10E-02	1.60E-09
		Total Activity	3.90E+00

DRILL **STAMPEDE User Supplied Information** DRILL
Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:24 **User Name:** Unit Vent Alert
Comments:

User Supplied Information

Meteorological Data Inputs:
Ground level wind velocity: **13.2 mi/hr**
Ground level wind from: **180 degrees**
User-selected Stability Class
Stability Class: **"D - Neutral"**

Monitored Unit Vent Release:
Unit Vent Release Rate entered: **2.50E+006 uCi/sec**

Reactor Shutdown Date/Time: **12/17/2013 14:24**
Release Start Date/Time: **12/17/2013 15:24**
Estimated Release Duration: **1.00 hours**

Nuclide Mixture: **Gap Inventory**

Calculated NOBLE GAS release rate: **1.19E+006 uCi/sec**

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	**uCi/sec**	**Nuclide**	**uCi/sec**	**Nuclide**	**uCi/sec**
Kr-83M:	2.53E+004	**I-131:**	3.12E+003	**Cs-134:**	1.05E+000
Kr-85:	1.05E+004	**I-132:**	3.18E+003	**Cs-137:**	8.25E-001
Kr-85M:	7.06E+004	**I-133:**	6.05E+003	**Ce/Pr-144:**	2.10E-004
Kr-87:	9.03E+004	**I-134:**	3.08E+003	**Ce-141:**	2.84E-004
Kr-88:	1.74E+005	**I-135:**	5.12E+003	**La-140:**	5.31E-004
Kr-89:	3.08E-001			**Mo-99:**	3.43E-001
Xe-131M:	3.12E+003			**Ru/Rh-106:**	8.25E-005
Xe-133:	6.22E+005			**Ru-103:**	2.50E-004
Xe-133M:	1.91E+004			**Sr/Y-90:**	9.10E-005
Xe-135:	1.45E+005			**Sr-89:**	1.82E-003
Xe-135M:	8.14E+003			**Te-132:**	1.35E-001
Xe-137:	9.53E+000			**Zr-95:**	3.13E-004
Xe-138:	2.66E+004				

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/17/2013 15:24 **User Name:** Unit Vent Alert
Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.436E-005
1.0	0:05	1.032E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.109E-008

Distance (miles)	Measurable Dose Rates: Immersion Whole Body noble gas gamma (rem/hr)	PAG Dose Rates: TEDE external + internal (rem/hr)	PAG Dose Rates: Iodine CDE Thyroid (rem/hr)
0.5	0.009	0.016	0.137
1.0	0.0033	0.006	0.051
2.0	0.001	0.002	0.018
5.0	0.000	0.001	0.004
7.5	0.000	0.000	0.002
10.0	0.000	0.000	0.001
20.0	0.000	0.000	0.000

Distance (miles)	Measurable Doses: Immersion Whole Body noble gas gamma (rem)	PAG Doses: TEDE external + internal (rem)	PAG Doses: Iodine CDE Thyroid (rem)
0.5	0.009	0.016	0.137
1.0	0.003	0.006	0.051
2.0	0.001	0.002	0.018
5.0	0.000	0.001	0.004
7.5	0.000	0.000	0.002
10.0	0.000	0.000	0.001
20.0	0.000	0.000	0.000

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 2 of 2

Calculations Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 13.2 mi/hr

Wind Direction: 180

Release Rate: 1.19E+006 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.006	0.002	0.001	0.000
CDE	0.051	0.018	0.004	0.001

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1

SHELTER IN PLACE ZONE(S): 2

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Dose Rate Projection of >3 mrem/hr (Immersion Whole Body Noble Gas Gamma) at the Site Boundary (1 Mile) for 15 minutes or longer the Emergency Classification Initiating Condition RA1 (ALERT) has been met.

PERFORMED BY: ______________________ 12/17/2013 3:24:44 PM

Name — Date/Time

REVIEWED BY: ______________________ ______________

Rad Manager/Radiological Director — Date/Time

12/17/2013 3:24:28 PM

DRILL **STAMPEDE User Supplied Information** DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/18/2013 07:54 **User Name:** Steam Line Site Alert
Comments:

User Supplied Information

Meteorological Data Inputs:
- **Ground level wind velocity:** 13.2 mi/hr
- **Ground level wind from:** 180 degrees
- **User-selected Stability Class**
- **Stability Class:** "D - Neutral"

Monitored S/G Tube Rupture Release:
Steam Activity: 4.50E+000 uCi/cc
Steam Flow Rate: 1.050 mlb/hr

Reactor Shutdown Date/Time: 12/18/2013 06:54
Release Start Date/Time: 12/18/2013 07:54
Estimated Release Duration: 1.00 hours

Nuclide Mixture: Noble Gas + Iodine
Iodine as percent of noble gas: 0.2%

Calculated NOBLE GAS release rate: 1.19E+007 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	**uCi/sec**	**Nuclide**	**uCi/sec**	**Nuclide**	**uCi/sec**
Kr-83M:	1.14E+004	I-131:	3.05E+003	Cs-134:	0.00E+000
Kr-85:	3.43E+005	I-132:	3.22E+003	Cs-137:	0.00E+000
Kr-85M:	5.79E+004	I-133:	4.88E+003	Ce/Pr-144:	0.00E+000
Kr-87:	2.55E+004	I-134:	4.22E+002	Ce-141:	0.00E+000
Kr-88:	9.89E+004	I-135:	1.23E+004	La-140:	0.00E+000
Kr-89:	4.25E-003			Mo-99:	0.00E+000
Xe-131M:	1.26E+005			Ru/Rh-106:	0.00E+000
Xe-133:	1.08E+007			Ru-103:	0.00E+000
Xe-133M:	1.87E+005			Sr/Y-90:	0.00E+000
Xe-135:	3.18E+005			Sr-89:	0.00E+000
Xe-135M:	1.23E+003			Te-132:	0.00E+000
Xe-137:	1.27E-001			Zr-95:	0.00E+000
Xe-138:	1.36E+003				

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/18/2013 07:54 **User Name:** Steam Line Site Alert
Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.436E-005
1.0	0:05	1.032E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.109E-008

	Measurable Dose Rates	PAG Dose Rates	
Distance (miles)	Immersion Whole Body noble gas gamma (rem/hr)	TEDE external + internal (rem/hr)	Iodine CDE Thyroid (rem/hr)
0.5	0.011	0.019	0.135
1.0	0.0042	0.007	0.050
2.0	0.002	0.003	0.017
5.0	0.000	0.001	0.004
7.5	0.000	0.000	0.002
10.0	0.000	0.000	0.001
20.0	0.000	0.000	0.000

	Measurable Doses	PAG Doses	
Distance (miles)	Immersion Whole Body noble gas gamma (rem)	TEDE external + internal (rem)	Iodine CDE Thyroid (rem)
0.5	0.011	0.019	0.135
1.0	0.004	0.007	0.050
2.0	0.002	0.003	0.017
5.0	0.000	0.001	0.004
7.5	0.000	0.000	0.002
10.0	0.000	0.000	0.001
20.0	0.000	0.000	0.000

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 2 of 2

Calculation: Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 13.2 mi/hr

Wind Direction: 180

Release Rate: 1.19E+007 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.007	0.003	0.001	0.000
CDE	0.050	0.017	0.004	0.001

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1

SHELTER IN PLACE ZONE(S): 2

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Dose Rate Projection of ≥ 3 mrem/hr (Immersion Whole Body Noble Gas Gamma) at the Site Boundary (1 Mile) for 15 minutes or longer the Emergency Classification Initiating Condition RA1 (ALERT) has been met.

PERFORMED BY: ______________________ 12/18/2013 7:55:14 AM

Name Date/Time

REVIEWED BY: ______________________ ______________

~~Rad Manager/Radiological Director~~ ~~Date/Time~~

DRILL **STAMPEDE User Supplied Information** DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:25 **User Name:** Unit Vent Site Area

Comments:

User Supplied Information

Meteorological Data Inputs:
Ground level wind velocity: 13.2 mi/hr
Ground level wind from: 180 degrees
User-selected Stability Class
Stability Class: "D - Neutral"

Monitored Unit Vent Release:
Unit Vent Release Rate entered: 2.50E+007 uCi/sec

Reactor Shutdown Date/Time: 12/17/2013 14:25
Release Start Date/Time: 12/17/2013 15:25
Estimated Release Duration: 1.00 hours

Nuclide Mixture: Gap Inventory

Calculated NOBLE GAS release rate: 1.19E+007 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	**uCi/sec**	**Nuclide**	**uCi/sec**	**Nuclide**	**uCi/sec**
Kr-83M:	2.52E+005	I-131:	3.12E+004	Cs-134:	1.05E+001
Kr-85:	1.05E+005	I-132:	3.18E+004	Cs-137:	8.24E+000
Kr-85M:	7.06E+005	I-133:	6.04E+004	Ce/Pr-144:	2.10E-003
Kr-87:	9.03E+005	I-134:	3.08E+004	Ce-141:	2.84E-003
Kr-88:	1.73E+006	I-135:	5.12E+004	La-140:	5.31E-003
Kr-89:	3.14E+000			Mo-99:	3.43E+000
Xe-131M:	3.12E+004			Ru/Rh-106:	8.24E-004
Xe-133:	6.22E+006			Ru-103:	2.50E-003
Xe-133M:	1.91E+005			Sr/Y-90:	9.10E-004
Xe-135:	1.45E+006			Sr-89:	1.82E-002
Xe-135M:	8.18E+004			Te-132:	1.35E+000
Xe-137:	9.74E+001			Zr-95:	3.13E-003
Xe-138:	2.67E+005				

DRILL STAMPEDE Results Information DRILL

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/17/2013 15:25 **User Name:** Unit Vent Sire Area
Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.436E-005
1.0	0:05	1.032E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.109E-008

Measurable Dose Rates / PAG Dose Rates

Distance (miles)	Immersion Whole Body noble gas gamma (rem/hr)	TEDE external + internal (rem/hr)	Iodine CDE Thyroid (rem/hr)
0.5	0.088	0.160	1.364
1.0	0.0332	0.060	0.510
2.0	0.012	0.021	0.176
5.0	0.003	0.005	0.041
7.5	0.002	0.003	0.021
10.0	0.001	0.002	0.014
20.0	0.000	0.001	0.005

Measurable Doses / PAG Doses

Distance (miles)	Immersion Whole Body noble gas gamma (rem)	TEDE external + internal (rem)	Iodine CDE Thyroid (rem)
0.5	0.088	0.160	1.364
1.0	0.033	0.060	0.510
2.0	0.012	0.021	0.176
5.0	0.003	0.005	0.041
7.5	0.002	0.003	0.021
10.0	0.001	0.002	0.014
20.0	0.000	0.001	0.005

12/17/2013 3:25:21 PM

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 2 of 2

Calculations Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 13.2 mi/hr

Wind Direction: 180

Release Rate: 1.19E+007 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.060	0.021	0.005	0.002
CDE	0.510	0.176	0.041	0.014

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1

SHELTER IN PLACE ZONE(S): 2

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Site Boundary (1 Mile) Dose Projection ≥ 0.1 rem TEDE and/or 0.5 rem Thyroid CDE the Emergency Classification Initiating Condition RS1 (SITE AREA EMERGENCY) has been met.

PERFORMED BY: ______________________ 12/17/2013 3:25:28 PM

Name Date/Time

REVIEWED BY: ______________________ ______________

Rad Manager/Radiological Director Date/Time

DRILL **STAMPEDE User Supplied Information** DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:28 User Name: Steam Line Site Area
Comments:

User Supplied Information

Meteorological Data Inputs:
Ground level wind velocity: 13.2 mi/hr
Ground level wind from: 180 degrees
User-selected Stability Class
Stability Class: "D - Neutral"

Monitored S/G Tube Rupture Release:
Steam Activity: 4.50E+001 uCi/cc
Steam Flow Rate: 1.050 mlb/hr

Reactor Shutdown Date/Time: 12/17/2013 14:28
Release Start Date/Time: 12/17/2013 15:28
Estimated Release Duration: 1.00 hours

Nuclide Mixture: Noble Gas + Iodine
Iodine as percent of noble gas: 0.2%

Calculated NOBLE GAS release rate: 1.20E+008 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	uCi/sec	Nuclide	uCi/sec	Nuclide	uCi/sec
Kr-83M:	1.14E+005	I-131:	3.07E+004	Cs-134:	0.00E+000
Kr-85:	3.45E+006	I-132:	3.23E+004	Cs-137:	0.00E+000
Kr-85M:	5.81E+005	I-133:	4.90E+004	Ce/Pr-144:	0.00E+000
Kr-87:	2.55E+005	I-134:	4.22E+003	Ce-141:	0.00E+000
Kr-88:	9.91E+005	I-135:	1.24E+005	La-140:	0.00E+000
Kr-89:	3.92E-002			Mo-99:	0.00E+000
Xe-131M:	1.27E+006			Ru/Rh-106:	0.00E+000
Xe-133:	1.08E+008			Ru-103:	0.00E+000
Xe-133M:	1.88E+006			Sr/Y-90:	0.00E+000
Xe-135:	3.19E+006			Sr-89:	0.00E+000
Xe-135M:	1.21E+004			Te-132:	0.00E+000
Xe-137:	1.19E+000			Zr-95:	0.00E+000
Xe-138:	1.34E+004				

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/17/2013 15:28 **User Name:** SteamLine Site Area
Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.436E-005
1.0	0:05	1.032E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.109E-008

	Measurable Dose Rates	PAG Dose Rates	
Distance (miles)	Immersion Whole Body noble gas gamma (rem/hr)	TEDE external + internal (rem/hr)	Iodine CDE Thyroid (rem/hr)
0.5	0.111	0.189	1.354
1.0	0.0423	0.072	0.506
2.0	0.015	0.025	0.175
5.0	0.004	0.006	0.041
7.5	0.002	0.003	0.021
10.0	0.001	0.002	0.013
20.0	0.001	0.001	0.005

	Measurable Doses	PAG Doses	
Distance (miles)	Immersion Whole Body noble gas gamma (rem)	TEDE external + internal (rem)	Iodine CDE Thyroid (rem)
0.5	0.111	0.189	1.354
1.0	0.042	0.072	0.506
2.0	0.015	0.025	0.175
5.0	0.004	0.006	0.041
7.5	0.002	0.003	0.021
10.0	0.001	0.002	0.013
20.0	0.001	0.001	0.005

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 2 of 2

Calculations Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 15.2 mi/hr

Wind Direction: 180

Release Rate: 1.20E+008 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.072	0.025	0.006	0.002
CDE	0.506	0.175	0.041	0.013

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1

SHELTER IN PLACE ZONE(S): 2

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Site Boundary (1 Mile) Dose Projection ≥ 0.1 rem TEDE and/or 0.5 rem Thyroid CDE the Emergency Classification Initiating Condition RS1 (SITE AREA EMERGENCY) has been met.

PERFORMED BY:

12/17/2013 3:29:00 PM

Name Date/Time

REVIEWED BY:

~~Rad Manager/Radiological Director~~ ~~Date/Time~~

12/17/2013 3:28:53 PM

DRILL **STAMPEDE User Supplied Information** DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:26 **User Name:** Unit Vent General
Comments:

User Supplied Information

Meteorological Data Inputs:
Ground level wind velocity: 13.2 mi/hr
Ground level wind from: 180 degrees
User-selected Stability Class
Stability Class: "D - Neutral"

Monitored Unit Vent Release:
Unit Vent Release Rate entered: 2.50E+008 uCi/sec

Reactor Shutdown Date/Time: 12/17/2013 14:26
Release Start Date/Time: 12/17/2013 15:26
Estimated Release Duration: 1.00 hours

Nuclide Mixture: Gap Inventory

Calculated NOBLE GAS release rate: 1.19E+008 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	**uCi/sec**	**Nuclide**	**uCi/sec**	**Nuclide**	**uCi/sec**
Kr-83M:	2.52E+006	I-131:	3.12E+005	Cs-134:	1.05E+002
Kr-85:	1.05E+006	I-132:	3.18E+005	Cs-137:	8.25E+001
Kr-85M:	7.06E+006	I-133:	6.04E+005	Ce/Pr-144:	2.10E-002
Kr-87:	9.03E+006	I-134:	3.08E+005	Ce-141:	2.84E-002
Kr-88:	1.73E+007	I-135:	5.12E+005	La-140:	5.31E-002
Kr-89:	3.10E+001			Mo-99:	3.43E+001
Xe-131M:	3.12E+005			Ru/Rh-106:	8.25E-003
Xe-133:	6.22E+007			Ru-103:	2.50E-002
Xe-133M:	1.91E+006			Sr/Y-90:	9.10E-003
Xe-135:	1.45E+007			Sr-89:	1.82E-001
Xe-135M:	8.16E+005			Te-132:	1.35E+001
Xe-137:	9.64E+002			Zr-95:	3.13E-002
Xe-138:	2.66E+006				

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/17/2013 15:26 **User Name:** Unit Vent General
Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0.02	2.686E-005	2.436E-005
1.0	0.05	1.032E-005	9.110E-006
2.0	0.09	3.755E-006	3.151E-006
5.0	0.23	1.004E-006	7.373E-007
7.5	0.34	5.704E-007	3.845E-007
10.0	0.45	3.851E-007	2.441E-007
20.0	1.31	1.541E-007	9.109E-008

	Measurable Dose Rates	PAG Dose Rates	
Distance (miles)	Immersion Whole Body noble gas gamma (rem/hr)	TEDE external + internal (rem/hr)	Iodine CDE Thyroid (rem/hr)
0.5	0.879	1.598	13.646
1.0	0.3322	0.601	5.099
2.0	0.117	0.210	1.762
5.0	0.029	0.050	0.411
7.5	0.016	0.027	0.214
10.0	0.010	0.017	0.135
20.0	0.003	0.006	0.050

	Measurable Doses	PAG Doses	
Distance (miles)	Immersion Whole Body noble gas gamma (rem)	TEDE external + internal (rem)	Iodine CDE Thyroid (rem)
0.5	0.879	1.598	13.646
1.0	0.332	0.601	5.099
2.0	0.117	0.210	1.762
5.0	0.029	0.050	0.411
7.5	0.016	0.027	0.214
10.0	0.010	0.017	0.135
20.0	0.003	0.006	0.050

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 2 of 2

Calculation: Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 13.2 mi/hr

Wind Direction: 180

Release Rate: 1.19E+008 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.601	0.210	0.050	0.017
CDE	$\boxed{5.099}$	1.762	0.411	0.135

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1, 2

SHELTER IN PLACE ZONE(S): 6, 11

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Site Boundary (1 Mile) Dose Projection ≥ 1 rem TEDE and/or 5 rem Thyroid CDE the Emergency Classification Initiating Condition RG1 (GENERAL EMERGENCY) has been met

PERFORMED BY:

Name	12/17/2013 3:26:33 PM Date/Time

REVIEWED BY:

Rad Manager/Radiological Director	Date/Time

12/17/2013 3:26:25 PM

DRILL STAMPEDE User Supplied Information DRILL

Revision 7.0.3.3 9/28/2011

Date/Time: 12/17/2013 15:30 User Name: SteamLine General
Comments:

User Supplied Information

Meteorological Data Inputs:
Ground level wind velocity: 13.2 mi/hr
Ground level wind from: 180 degrees
User-selected Stability Class
Stability Class: "D - Neutral"

Monitored S/G Tube Rupture Release:
Steam Activity: 4.50E+002 uCi/cc
Steam Flow Rate: 1.050 mlb/hr

Reactor Shutdown Date/Time: 12/17/2013 14:30
Release Start Date/Time: 12/17/2013 15:30
Estimated Release Duration: 1.00 hours

Nuclide Mixture: Noble Gas + Iodine
Iodine as percent of noble gas: 0.2%

Calculated NOBLE GAS release rate: 1.20E+009 uCi/sec

NOBLE GAS		IODINE		PARTICULATE	
Nuclide	uCi/sec	Nuclide	uCi/sec	Nuclide	uCi/sec
Kr-83M:	1.14E+006	I-131:	3.07E+005	Cs-134:	0.00E+000
Kr-85:	3.45E+007	I-132:	3.24E+005	Cs-137:	0.00E+000
Kr-85M:	5.82E+006	I-133:	4.90E+005	Ce/Pr-144:	0.00E+000
Kr-87:	2.56E+006	I-134:	4.24E+004	Ce-141:	0.00E+000
Kr-88:	9.92E+006	I-135:	1.24E+006	La-140:	0.00E+000
Kr-89:	4.13E-001			Mo-99:	0.00E+000
Xe-131M:	1.27E+007			Ru/Rh-106:	0.00E+000
Xe-133:	1.08E+009			Ru-103:	0.00E+000
Xe-133M:	1.88E+007			Sr/Y-90:	0.00E+000
Xe-135:	3.19E+007			Sr-89:	0.00E+000
Xe-135M:	1.23E+005			Te-132:	0.00E+000
Xe-137:	1.24E+001			Zr-95:	0.00E+000
Xe-138:	1.35E+005				

DRILL **STAMPEDE Results Information** DRILL

Revision 7.0.3.3 9/28/2011 Page 1 of 2

Date/Time: 12/17/2013 15:30 **User Name:** Steam Line General
Comments:

Plume Information

Distance (miles)	Plume Travel Time (hours:minutes)	CHI/Q Value (sec/m³)	CHI/Q DEPL (sec/m³)
0.5	0:02	2.686E-005	2.436E-005
1.0	0:05	1.032E-005	9.110E-006
2.0	0:09	3.755E-006	3.151E-006
5.0	0:23	1.004E-006	7.373E-007
7.5	0:34	5.704E-007	3.845E-007
10.0	0:45	3.851E-007	2.441E-007
20.0	1:31	1.541E-007	9.109E-008

	Measurable Dose Rates	PAG Dose Rates	
Distance (miles)	Immersion Whole Body noble gas gamma (rem/hr)	TEDE external + internal (rem/hr)	Iodine CDE Thyroid (rem/hr)
0.5	1.108	1.895	13.537
1.0	0.4236	0.717	5.058
2.0	0.153	0.254	1.747
5.0	0.040	0.063	0.407
7.5	0.022	0.034	0.211
10.0	0.015	0.022	0.134
20.0	0.006	0.008	0.049

	Measurable Doses	PAG Doses	
Distance (miles)	Immersion Whole Body noble gas gamma (rem)	TEDE external + internal (rem)	Iodine CDE Thyroid (rem)
0.5	1.108	1.895	13.537
1.0	0.424	0.717	5.058
2.0	0.153	0.254	1.747
5.0	0.040	0.063	0.407
7.5	0.022	0.034	0.211
10.0	0.015	0.022	0.134
20.0	0.006	0.008	0.049

DRILL **STAMPEDE Results Information** DRILL

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Calculation: Completed

RESULTS

Method of Projection: STAMPEDE

Wind Velocity: 13.2 mi/hr

Wind Direction: 180

Release Rate: 1.20E+009 uCi/sec

Offsite Dose Projection (rem):

	1 mile	2 miles	5 miles	10 miles
TEDE	0.727	0.254	0.063	0.022
CDE	5.058	1.747	0.407	0.134

Projected duration of release: 1.0 hours

A General Emergency Requires a Protective Action Recommendation

EVACUATE ZONE(S): 1, 2

SHELTER IN PLACE ZONE(S): 6, 11

AFFECTED DOWNWIND SECTORS: R, A, B

All Remaining Zones Go Indoors And Monitor EAS Radio Station

Based on a Site Boundary (1 Mile) Dose Projection ≥ 1 rem TEDE and/or 5 rem Thyroid CDE the Emergency Classification Initiating Condition RG1 (GENERAL EMERGENCY) has been met

PERFORMED BY: ______________________ 12/17/2013 3:30:48 PM

Name Date/Time

REVIEWED BY: ______________________ ______________

~~Rad Manager/Radiological Director~~ ~~Date/Time~~

12/17/2013 3:30:39 PM

3.0 Alarm Set point Adjustments

3.1 Liquid Effluents

3.1.1 Control Requirements

Control 3/4.11.1.1 of Part A of the ODCM requires that the concentration of radioactive material released at any time from the South Texas Project (STP) to unrestricted areas be limited to ten times the Effluent Concentration (ECs) in water. The ECs are as indicated in 10CFR20, Appendix B, Table 2, Column 2 for nuclides other than dissolved or entrained noble gases. Noble gas concentrations must be limited to 2.0E-04 uCi/ml.

3.1.2 Interpretation

Liquid effluent discharges from STP are diluted by a 7000-acre reservoir. Plant discharges are all routed into the cooling reservoir where substantial dilution and radioactive decay may occur before ultimate release from the site. The reservoir lies totally within the confines of the site and the use of its water is restricted to plant operation. Recreational use of the reservoir is limited to occasional catch and release fishing tournaments for employees and their families. This recreational use is closely controlled to prevent ingestion of radioactive effluents. Liquid effluents diluted into the cooling reservoir may be released during:

a) scheduled blowdown operations to the Colorado River,

b) passive hydraulic relief well flow,

c) dilution into the shallow ground water aquifer, or

d) spillway releases.

The blowdown releases will be planned; however, the other releases are not controlled by the operations staff. To assure that the provisions of Part A, Control 3/4.11.1.1 are satisfied, the concentrations of radionuclides in the reservoir shall be maintained at levels less than ten times the limits of 10CFR20, Appendix B, Table 2, Column 2. Hence, STP shall apply controls on the concentration of liquid effluents as they are discharged into the reservoir to assure that any releases to uncontrolled areas from the reservoir meet the requirements of Control 3/4.11.1.1.

3.1.3 Implementation

Concentrations of radionuclides in the cooling reservoir will be controlled such that the sum of their ratios to the ECs, A, remains less than ten as indicated in Equation 3.1a below:

$$A = \frac{C_1}{EC_1} + \frac{C_2}{EC_2} + \ldots.. + \frac{C_i}{EC_i} < 10 \qquad \text{Eq.3.1a}$$

Where $C_1, C_2, \ldots, C_i$ are the measured nuclide concentrations of a representative sample of reservoir water (uCi/ml);

$EC_1, EC_2, \ldots, EC_i$ are the associated effluent concentrations of those nuclides which collectively contribute at least 90% to the total dose.

As long as "**A**" from equation 3.1a above is less than ten, releases from the reservoir to the off-site environment will meet the requirements of Control 3/4.11.1.1. In order to assure that **A** never exceeds ten, the dilution afforded by the circulating coolant and auxiliary cooling water flows must be estimated. The dilution of liquid radioactive waste discharges into the circulating coolant from each unit is calculated as indicated below:

$$A = [DF_r * A_r] + [DF_c * A_c] \qquad \text{Eq.3.1b}$$

$$DF_r = \frac{F_r}{F_c + F_r} \qquad \text{Eq.3.1c}$$

$$DF_c = \frac{F_c}{F_c + F_r} \qquad \text{Eq.3.1d}$$

where:

- A = the sum of the effluent concentrations in the circulating coolant as it reenters the reservoir divided by their ECs; A < 10.
- DF_r = dilution factor for a radioactive waste
- A_r = number of ECs permitted in the radioactive waste flow from the waste monitor tank, unit less factor;
- DF_c = dilution factor for circulating coolant
- A_c = number of ECs in the circulating coolant before addition of the radioactive waste stream as measured periodically for the reservoir, unit less factor;
- F_r = average flow rate of radioactive waste as determined by the rated pump capacity of the radioactive waste discharge, gal/min;
- F_c = flow rate of circulating coolant and the open loop auxiliary cooling water, normally 4.5E5 gal/min (4.5E5 is 1/2 the normal circulating coolant flow of each unit since liquid radioactive waste is discharged into only one of two 138" lines). F_c may be determined by multiplying the number of circulating coolant pumps operating by the rated pump capacity;

Since liquid effluents are released as batches, the very large dilution factor afforded by the reservoir would not be fully used even if high concentrations of liquid radioactive waste were infrequently discharged from the plant. As an operational rule, liquid effluents should not be discharged to the reservoir if the value of **A**, as described by Eq. 3.1a and as calculated by
Eq. 3.1b, exceeds "ten". From practical experience, limiting liquid effluent discharges such that A≤ 10 maintains the measured reservoir concentrations within the limits of 10CFR20, Table 2, column 2.

If the value of "A" in equation 3.1b is set to its limiting value of 10, the terms in Eq. 3.1b above can be rearranged as shown below:

$$Ar = \frac{[F_c * (10 - A_c)]}{F_r} + 10 \qquad \text{Eq.3.1e}$$

An estimate of A_r appropriate for limiting routine batch discharges to the reservoir can be made assuming that the radioactive waste stream flow is at its nominal value, the flow of dilution water is at its minimum, and that the reservoir is virtually unpolluted. In this case the values for each variable above become:

F_c = 113,000 gpm (one circulating coolant water pump)

F_r = 250 gpm (nominal flow rate limit for radioactive waste discharge pump);

A_c = 0 (reflecting good radioactive discharge management)

Hence, Eq. 3.1e can be solved for A_r as:

A_r = [113,000/250 * (10-0)] + 10 = 4530

This suggests that for normal operation with a "clean" reservoir, the administrative limit for discharges should limit discharge concentrations to no more than about 4530 times the effective EC of the radioactive waste stream.

The radioactive waste stream itself is characterized by a mixture of radionuclides at concentrations C_1, C_2, ..., C_j. The effective EC of this waste stream can be estimated from the radiochemical analysis of the waste monitor tank prior to a batch discharge using the following formula for effective EC:

$$EC_{eff} = \frac{\Sigma C_j}{\Sigma (C_j/EC_j)} \qquad \text{Eq.3.1f}$$

where

C_j = concentrations of individual radionuclides, "j", in the mixture, uCi/ml

$\sum C_j$ = sum of the concentrations in the waste monitor tank, uCi/ml

EC_j = effluent concentrations listed in 10CFR20, Appendix B, Table 2, column 2, for each radionuclide, "j", uCi/ml

EC_{eff} = effective EC for a mixture of radionuclides, uCi/ml

The limiting concentration, LC, may be estimated by multiplying the value of EC_{eff} from Eq. 3.1f by the factor A_r from Eq. 3.1e.

$$LC = A_r * EC_{eff} \qquad \text{Eq. 3.1g}$$

This limiting concentration could be used as the basis for setting the liquid effluent monitor, RT-8038, if the instrument could detect these nuclides. However, the model RD-53 detector used in the RT-8038 monitor is sensitive to only gamma emitting nuclides, and its sensitivity to individual gamma emitters is not the same. The alarm set point must be based on the response of the RD-53 detector in counts per minute, cpm, to a nuclide mix in a particular discharge corresponding to an LC.

The count rate corresponding to the effective effluent concentration, CR, can be calculated in a manner similar to the methods of equation 3.1f.

$$CR = \frac{\Sigma (C_j * Er_j)}{\Sigma (C_j/EC_j)} \qquad \text{Eq.3.1h}$$

where

CR = count rate, cpm, associated with one EC_{eff}

Er_j = RD-53 response to nuclide "j", (cpm)/(uCi/ml)

The limiting count rate, LCR, may be estimated as was the limiting concentration in equation 3.1g.

$$LCR = A_r * CR \qquad \text{Eq. 3.1i}$$

The following example uses the average mixture of radionuclides measured in the liquid effluent released during August 1988 to calculate the limiting concentration and corresponding limiting count rate for the RT-8038 monitor:

Nuclide	Concentration C (uCi/ml)	EC (uCi/ml)	Concentration/EC (C/EC)	Er (cpm)/(uCi/ml)	C * Er (cpm)
H-3	1.74E-02	1E-03	1.7E+01	0	0
Cr-51	4.22E-08	5E-04	8.4E-05	1.45E+07	6.12E-01
Mn-54	2.80E-08	3E-05	9.3E-04	1.40E+08	5.91E+00
Co-58	1.01E-06	2E-05	5.1E-02	1.83E+08	1.85E+02
Zr-95	3.41E-08	2E-05	1.7E-03	1.40E+08	4.77E+00
Nb-95	3.41E-08	3E-05	1.1E-03	1.40E+08	4.77E+00
Co-60	2.20E-08	3E-06	7.3E-03	2.65E+08	5.83E+00
Xe-133	3.96E-05	2E-04	2.0E-01	0	0
Xe-135	2.48E-07	2E-04	1.2E-03	1.31E+08	3.25E+01
	1.74E-02		1.7E+01		2.39E+02

$$EC_{eff} = (\Sigma\ C_j)/(\Sigma (C_j/EC_j)) = (1.74E-02\ uCi/ml)/1.7E+01$$
$$= 1.0E-03\ uCi/ml$$

$$CR = (\Sigma (C_j * Er_j))\ /\ (\Sigma (C_j/EC_j)) = (2.39E+02\ cpm)/1.7E+01$$
$$= 1.4E+01\ cpm$$

The limiting discharge concentration in this example can be estimated using Eq. 3.1g as shown below:

LC = 4530 * 1.0E-03 uCi/ml = 4.5 uCi/ml

Note that radionuclides were included in the calculation which could not be detected by the model RD-53 detector. Examples of such nuclides include H-3, C-14, Fe-55, Tc-99, and Sr-90.

The alarm set point must be calculated based on the count rate RT-8038 would indicate if this limiting concentration were present. This count rate can be estimated using Eq. 3.1i as shown below:

LCR = 4530 * 14 cpm = 63,000 cpm

Note that no provision was made for the detector background, uncertainty in instrument response, or any safety factor in this calculation. Plant implementing procedures shall provide instructions for inclusion of background in the set point estimation and shall have provisions for cleaning the detector if the background becomes large enough to interfere with measurements.

As a result of improved radioactive waste treatment the gamma emitter concentrations of radioactive liquid effluents may be a very small fraction of the non-gamma emitter concentrations. The limiting count rate may yield an alarm set point lower than the expected count rate or expected monitor response. In these cases, plant implementing procedures may provide instructions for determining the alarm set point.

The limiting count rate calculated in Eq. 3.1i above should include these final adjustments as shown below to yield the alarm set point:

alarm set point = (LCR) * SF + bkg   Eq. 3.1j

where

SF = safety factor which includes the error margin calculated for this monitor. The effluent monitors are assumed to be accurate to 25%. An appropriate safety factor therefore should be set at 0.75 to reasonably assure an alarm and automatic discharge termination at or before exceeding the limiting concentration. The reader should note that the limiting concentration is calculated at the monitor, before the vast dilution provided by the reservoir. Hence, even if the LC were substantially exceeded for discharges into the reservoir, little chance exists to exceed an EC in unrestricted areas.

bkg = detector background in cpm

For the example chosen above and assuming bkg = 0, this calculation would look like:

alarm set point= 63,000 cpm * 0.75 + 0 cpm
= 47,000 cpm

The detector response function is not as precisely known as this example would suggest; hence, 20-30% differences between estimated alarm set points are not significant. It may be convenient for this alarm set point to be expressed in units of uCi/ml or uCi/sec based on the appropriate uCi/cpm conversion factor. The RT-8038 alarm may be set to a default value if the default does not exceed the value calculated in Eq. 3.1j above.

The alarm set point and calibration factors for liquid effluent monitor RT-8038 are applied to batch discharges and are adjusted for each discharge if the nuclide mix is sufficiently different to change either the discharge limit or calibration factor from the previous setting by more than 25%. If the alarm set point is exceeded during a batch discharge, the discharge is automatically terminated until the batch discharge activity is confirmed. Discharges from two or more waste monitor tanks from a single unit simultaneously are prohibited. Hence, this ODCM does not provide instructions for simultaneous discharges from the radioactive waste monitoring system.

RT-8038 is the only liquid effluent monitor for each unit. Gamma detection instrumentation is installed for other systems (RT-8041 and RT-8042) as shown in Figure B2-2. These process control instruments have alarm set points at 1.0E-06 uCi/ml or less (one EC for Cs-137) above background and act to identify rather than to quantify activity in systems during a discharge. If activity is identified, it is sampled and discharged (if treatment is not required to meet the limiting concentration of equation 3.1g) or is routed to the liquid waste processing system for treatment and discharge as a routine liquid effluent.

3.2 Gaseous Effluents

3.2.1 Control Requirements

Control 3/4.11.2.1 of Part A of the ODCM requires that the dose rates at the site boundary and beyond from noble gases be no greater than 500 mrem/year total body and 3000 mrem/year to the skin. Furthermore, dose rates due to I-131, I-133, H-3, and all radionuclides in particulate form with half-lives greater than eight days shall be less than or equal to 1500 mrem/year to any organ.

3.2.2 Interpretation

In order to help ensure that these limits are not exceeded, the alarm set points for the MAB/RCB common exhaust noble gas monitors are to be calculated such that the nearest offsite receptor would not be exposed to noble gas concentrations likely to produce a dose rate greater than Control 3/4.11.2.1 from the combined releases from Units 1 and 2. Iodines, tritium, and all other radionuclides contributing to organ doses are not considered for purposes of setting alarm points since they are sampled and not monitored.

3.2.3 Implementation

The nearest site boundary is about a mile from either unit; hence, a factor to relate the release to the concentration at the site boundary is necessary. UFSAR Tables 2.3-25 and 2.3-27 contain 2-hour and annual average X/Q values at the site boundary in each of 16 sectors. Logarithmic interpolation provides an estimate of 5.3E-06(sec/m^3) for the 500 hour X/Q in the NNW sector. This value of X/Q shall be used to provide estimates of dilution for the purpose of setting alarm points for routine releases.

The most prevalent radioactive gas present in the effluent may be used to control emissions when the noble gas effluent is dominated by a single nuclide. If no single nuclide dominates, then release alarm set points should be based on the average mixture found.

The dose rate to individuals at the site boundary may be estimated using the equations of section B4.4.2 (Eq. 4.4d for whole body dose rate and Eq. 4.4e for skin dose rate). Therefore, the limits of Control 3/4.11.2.1 may be expressed in terms of the following equations for each noble gas:

$$\text{whole body dose rate} = Dr_{gamma} * 8760 < 0.5 \text{ rem/yr} \qquad \text{Eq. 3.2a}$$

$$\text{skin dose rate} = Dr_{skin} * 8760 < 3 \text{ rem/yr} \qquad \text{Eq. 3.2b}$$

where 8760 = units conversion factor (hr/yr)

Dr_{gamma} = whole body dose rate, rem/hr

Dr_{skin} = skin dose rate, rem/hr

$$Dr_{gamma} = 0.114 * X/Q * \sum_i (Qi * Dfi_{gamma}) * Sf \qquad \text{(rem/hr)}$$

$$Dr_{skin} = 0.111 * S_f * Dr_{gramma(air)} + Dr_{beta(skin)} \qquad \text{(rem/h)}$$

and where

$$Dr_{gamma(air)} = 0.114 * X/Q * \sum_i (Qi * Dfi_{gamma(air)}) \qquad \text{(rad/h)}$$

$$Dr_{beta(skin)} = 0.114 * X/Q * \sum_i (Qi * Dfi_{beta(skin)}) \qquad \text{(rem/h)}$$

Dfi_{gamma}	=	gamma dose to tissue conversion factor by nuclide from Table B-1, Regulatory Guide 1.109 (mrem-m^3/pCi-yr),
$Dfi_{gamma(air)}$	=	gamma dose to air conversion factor by nuclide from Table B-1, Regulatory Guide 1.109 (mrad-m^3/pCi-yr),
$Dfi_{beta(skin)}$	=	beta dose to tissue conversion factor by nuclide from Table B-1, Regulatory Guide 1.109 (mrem-m^3/pCi-yr),
1.11	=	ratio of the mass stopping powers for electrons in air to tissue.
0.114	=	conversion factor from (mrem-m^3)/(pCi-yr) to (rem-m^3)/(uCi-hr)
Qi	=	isotope "i" release rate (uCi/sec) from monitors RT-8010B
X/Q	=	5.3E-06 (sec/m^3);
S_f	=	1.0 (a shielding factor set to one since it is not applicable for instantaneous dose rate calculations);

Hence, release rates, Q_j, can be calculated for each noble gas which would correspond to the whole body (0.5 rem/yr) and skin (3 rem/yr) limits of Eqs. 3.2a and 3.2b. Furthermore, if the release rate is divided by the unit vent flow rate, the limiting stack concentration may be estimated for each noble gas as indicated below and as listed in Table B3-3:

$$(\text{limiting stack concentrations})_{wb} = LC_{wb} = Q_j / F \qquad \text{Eq. 3.2c}$$

$$= \frac{94.5}{Dfi_{gamma} * F} \text{ (uCi/cc)}$$

$$(\text{limiting stack concentrations})_{skin} = LC_{skin} = Q_j / F \qquad \text{Eq. 3.2d}$$

$$= \frac{567}{(1.1 * Dfi_{gamma} + Dfi_{beta(skin)}) * F} \text{ (uCi/cc)}$$

where F = unit vent flow rate (200,000 scfm = 9.4E+07 cc/sec)
Qj = solved from the Eq. 3.2a and 3.2b with Dr_{gamma} = 0.5 rem/yr and Dr_{skin} = 3 rem/yr
94.5 and 567 have units of (mrem-µCi-m^3)/(pCi-yr-sec)

As for the liquid monitor, a safety factor should be included to afford operators an opportunity to take corrective action should a release threaten to exceed the Control limit. However, an allocation factor is also necessary to assure that the off-site dose rate due to effluents from other potential release points do not combine to exceed the Control limit. Errors associated with the effluent monitoring must also be considered in estimating the set point. Lastly, the detector background should be included in the alarm set point calculation. The set point calculation should therefore resemble Eq. 3.2e as shown below:

alarm set point = [(LC) * SF * AF] + bkg   Eq. 3.2e

where LC = either the whole body or skin limiting stack concentration, whichever is less, uCi/cc

SF = safety factor which includes a margin for monitor error for this monitor (Bechtel calculation 9ZC6008 documents the RD-52 detector statistical accuracy to be about 40%. Hence, the safety factor is estimated as: 1 - 0.4 = 0.6). Measurements of grab samples taken during noble gas releases has demonstrated that the RD-52 detector is more accurate than the engineering calculation suggests. Thus the safety factor of 0.6 is conservative.

AF = allocation factor (ex: 0.5 or half for each unit)

bkg = detector background, uCi/cc

EXAMPLE CALCULATION

The routine release point alarm setting should be limited to the value listed for Xe-133 in Table B3-3. However, a calculation for a release with several noble gases could be made as shown below if a very precise estimate of the limit were necessary.

Given:

Nuclide	Measured Concentration, C (uCi/cc)	Limiting Concentration, LC (uCi/cc)		C / LC	
		Whole Body	Skin	Whole Body	Skin
Ar-41	1.0E-06	1.14E-04	4.63E-04	8.77E-03	2.16E-03
Kr-85	1.0E-06	6.24E-02	4.44E-03	1.6E-05	2.25E-04
Xe-133	4.0E-05	3.42E-03	8.64E-03	1.17E-02	4.63E-03
				2.05E-02	7.02E-03

The fraction of the limiting concentration for both whole body and skin exposures is estimated as the sum of the ratios of the measured release concentrations divided by the corresponding limiting concentrations from Table B3-3. These values are listed in the table above under the column "C/LC." In this example, the sum for the whole body exposure is more limiting than for the skin (normal result). This sum represents the fraction of the limiting concentration for the current release. The limiting concentration for each nuclide in the mixture could be increased by the factor listed in the column "C/∑(C/LC)" below:

C/∑ (C/LC)	**Re_i**	**(C/∑ (C/LC))*Re_i**
4.88E-05	2.6	1.26E-04
4.88E-05	2.4	1.17E-04
1.95E-03	1.0	1.95E-03
2.04E-03		LC_{eff} = 2.19E-03

Since the monitor does not respond to all radionuclides the same, the product of value "C/∑(C/LC)" and "Re_i" (the relative response from Table B3-2) yields the monitor response to each nuclide in the mixture at their respective maximum concentrations, column "C/∑(C/LC) * Re_i". The sum of these concentrations, LC_{eff}, is the effective limiting concentration indicated at the monitor when the whole body or skin dose rate at the site boundary equals 500 mrem/yr or 3000 mrem/yr, respectively.

$$LCeff = \sum_i \left((Ci / (Ci / LCi)) * Rei \right) \qquad \text{Eq.3.2f}$$

The alarm set point would be estimated in accordance with Equation 3.2e as shown below where LC_{eff} is used in place of LC:

alarm set point = [2.19E-03 uCi/cc * 0.5 * 0.6] + 0 uCi/cc
= 6.6E-05 uCi/cc or 67,000 uCi/sec (at a vent flow rate of 200,000 cfm)

The alert set point may be chosen at any value, but typically might be set at about 80% of the alarm limit.

Note that the limiting release concentration (2.04E-03 uCi/cc) is about 93% of the indicated limiting concentration (2.18E-03 uCi/cc) in this example because ^{41}Ar and ^{85}Kr do not have the same monitor response as ^{133}Xe to which the detector is calibrated.

If the alarm set point calculated using this method is too conservative to permit a short term release, the set point may be recalculated using the anticipated X/Q during the release period using the best available forecast data and Equation 4.4d of Section B4.4. If no concurrent release from Unit 2 is projected, the allocation factor in Equation 3.2e could be increased to unity if the release were closely monitored. Equation 4.4d used to calculate the sector average X/Q would not provide conservative X/Q estimates and, hence, the release would require close monitoring to assure compliance with the Control limit.

Some process control monitors exist within the plant which are used to limit the effluent from particular parts of the plant should they threaten to cause the unit vent monitor to exceed its alarm set point. Although these process monitor set points are not required to be set in accordance with the ODCM, these alarm set points could be related to the unit vent alarm set point based on their contribution to the unit vent exhaust rate. For example, the containment supplemental purge line could have its set point calculated as:

$$(\text{alarm})_{\text{purge}} = \frac{\text{unit vent flow}}{\text{supp. purge flow}} * (\text{Unit Vent alarm setting}) * \text{AF} \qquad \text{Eq.3.2g}$$

where unit vent flow rate = 200,000 scfm = 94 m^3/sec
supp. purge flow = 5,000 scfm = 2.4 m^3/sec
Unit Vent alarm setting = current unit vent alarm set point
AF' = additional allocation factor (note: the sum of all allocation factors shall be 1.0)

For example: 0.2 for supplemental purge
0.2 for purge line
0.2 for fuel handling building
0.2 for waste gas process system
0.2 for remainder of plant

Although Control 3/4.11.2.1 requires periodic confirmation that the off-site dose rates calculated for particulates, tritium, and iodine do not exceed 1500 mrem/year to any organ, alarm/trip set points are not practicable to apply when considering instantaneous iodine and particulate dose rates. NUREG-0133 acknowledges that for practical reasons such alarm set points could not be set unambiguously.

Although the above method is suitable for the common MAB/RCB exhaust system, two other monitored atmospheric exhausts are not addressed. The condenser vacuum pumps may exhaust to the roof of TGB or to the unit vent. This alarm set point is dictated by plant safety considerations and is more conservative than off-site dose criteria. The flow (dry gas) through this exhaust is only about 2 (cubic meters/minute) and hence would not contribute significantly to the off-site dose unless the concentration of noble gas was exceedingly high, higher in fact than levels STP would permit to be exhausted onto the top level of the turbine building. The set point for this detector is adjusted to assure the safety of plant personnel if exhaust is to the TGB roof. Any releases from this exhaust whether routed to the unit vent or not will be included in monthly off-site dose calculations and will be reported in conformance with Regulatory Guide 1.21.

The other potential release is through the main atmospheric steam dumps which may release activity contained in the secondary coolant following turbine trips at greater than 50% power. These events are not frequent and the radiation monitoring system is not capable of accurately measuring this type of release. The Annual Effluent Release Report will contain estimates for such releases based on the measured nuclide concentrations in the secondary coolant and the estimated mass of coolant vented. For example:

release of nuclide "i" = Flowrate * Time * $Concentration_i$

where	Flowrate	= estimated steam vent rate, lbs/sec
	Time	= duration of release, sec
	$Concentration_i$	= concentration of nuclide "i", uCi/lbs.

Plant operation with the RT-8010B alarm set using the methods of this section and with the 500 hour X/Q shall demonstrate that the off-site dose rate does not exceed the Control 3/4.11.2.1 limits. If an unusual operating situation arises such that the release rate approaches or exceeds the RT-8010B alarm set point, the actual dose rate shall be calculated using actual meteorological and release data with the methods of ODCM Part B, Section 4.3. The real time dose rate may be used to demonstrate compliance with Control 3.11.2.1.

Table B3-1: Liquid Release Detector, RD-53, Response to
1 uCi/ml of Each Nuclide

Count Rate	
Nuclide (uCi/ml)	**Response (Er) (cpm)/(uCi/ml)**
Be-7	1.50E+07
Sc-46	2.74E+08
Cr-51	1.45E+07
Mn-54	1.40E+08
Co-57	9.78E+07
Co-58	1.83E+08
Fe-59	1.38E+08
Co-60	2.65E+08
Zn-65	7.26E+07
Kr-85	6.24E+07
Kr-85m	1.07E+08
Rb-86	1.18E+07
Kr-87	8.86E+07
Kr-88	8.49E+07
Sr-91	1.90E+08
Zr-95	1.40E+08
Nb-95	1.40E+08
Zr-97	1.64E+08
Nb-97	1.42E+08
Mo-99	2.74E+08
Tc-99m	9.82E+07
Ag-110m	4.38E+08
Sn-113	9.86E+07
Sb-122	1.09E+08
Sb-124	2.49E+08
Sb-125	1.21E+08
Te-129m	4.12E+06
I-130	4.72E+08
Xe-131m	2.35E+06
I-131	1.43E+08
Te-131m	2.48E+08
Te-132	1.19E+08
Xe-133	0
Xe-133m	1.41E+07
I-133	1.45E+08
Cs-134	3.17E+08
Xe-135	1.31E+08
Xe-135m	1.16E+08
I-135	1.79E+08
Cs-136	3.90E+08
Cs-137	1.21E+08
Xe-138	1.24E+08
Ba-140	4.65E+07
La-140	2.74E+08
Ce-144	1.13E+07
Hf-181	2.00E+08
W-187	1.11E+08

Table B3-1: Liquid Release Detector, RD-53, Response to
1 uCi/ml of Each Nuclide
(Continued)

The response of the RD-53 detectors to different radionuclides can be estimated using the gamma emissions from each radionuclide and the monitor's most recent calibration data (detection efficiencies used in this example are from Figure B2-5). The estimated response values listed above were estimated as shown below:

$$Er = \frac{\text{detected cpm}}{\text{uCi/ml of nuclide}} = Eff_1 * n_1 + Eff_2 * n_2 + ... + Eff_i * ni$$

where Eff_i = gamma detection efficiency for each gamma of energy class "i" from Figure B2-5 (cpm)/(uCi/ml)),

n_i = frequency of gamma energy class "i" emission per decay.

Pure beta emitters and alpha emitters produce zero response on this instrument. Gamma emitters with energies less than 100 keV should produce little or no response on this monitor.

<u>Example Calculations for Entrained Noble Gases</u>

Nuclide	Detection		Gamma Fraction	Er(cpm)/(uCi/ml)
	Gamma Energy(keV)	Efficiency (cpm)/(uCi/ml)		
Kr-85m	151	1.15×10^8	0.755	8.68E+07
	304	1.46×10^8	0.140	2.04E+07
				Total =1.07E+06
Xe-131m	164	1.20×10^8	0.0196	2.35E+06
Xe-133	81	0	0.371	0
Xe-133m	233	1.37×10^8	0.103	1.41E+07
Xe-135	250	1.40×10^8	0.903	1.264E+08
	608	1.44×10^8	0.0291	4.2E+06
				Total =1.31E+08
Xe-135m	527	1.45×10^8	0.800	1.16E+08

Table B3-2: Noble Gas Detector, RD-52, Response to 1 uCi/cc of Each Nuclide

Nuclide	Count Rate Response (E) cpm / uCi/cc	Indicated Response (Re_i) uCi/cc (Xe-133 Equivalent) / uCi/cc
Ar-41	9.4E+07	2.6
Kr-85m	6.9E+07	1.9
Kr-85	8.55E+07	2.4
Kr-87	9.9E+07	2.8
Kr-88	8.3E+07	2.3
Kr-89	1.0E+08	2.8
Kr-90	1.0E+08	2.8
Xe-131m	5.5E+05	0.015
Xe-133m	4.8E+05	0.14
Xe-133	3.55E+07	1.0
Xe-135m	1.5E+07	0.042
Xe-135	8.9E+07	2.5
Xe-137	1.0E+08	2.8
Xe-138	1.0E+08	2.8

The RD-52 beta radiation detectors are used in the RT-8010B gaseous radioactive effluent discharge monitor. The response of the detector to different radionuclides can be estimated using the beta emissions from each radionuclide and the monitor's most recent calibration (beta detection efficiencies used in this example are from Figure B2-3). The response values in the column labeled "Count Rate Response (E)" were calculated as shown below:

$$E = \text{detector cpm/(uCi/cc)} = Eff_1 * n_1 + Eff_2 + n_2 + \ldots + Eff_i * n_i$$

Where Eff_i = beta detection efficiency each beta of energy class "i" from Figure B2-3 (cpm per uCi/cc),

n_i = frequency of beta energy class "i" emission per decay.

The efficiency of detection factor relative to Xe-133, Re_i, may be calculated from the above efficiency as follows:

$$Rei = E / \frac{cpm}{uCi/cc} \text{ of reference nuclide}$$

The reference nuclide is the radionuclide with which the detector was calibrated and the one for which 1 uCi/cc indicated by the monitor actually corresponds to 1 uCi/cc in the sample line. Most other radionuclides will only approximately reflect a 1 uCi/cc monitor response when 1 uCi/cc is in the sample line. Thus, the "Indicated Detector Response (Re_i)" column shows how well the RT-8010B monitor estimates the concentrations of each radionuclide potentially in the gaseous effluent stream.

Example Calculations for Noble Gas Releases

Nuclide	Beta Energy max (keV)	Detection Efficiency (cpm)/(uCi/ml)	Beta Fraction	E (cpm)/(uCi/cc)
Ar-41	1200	9.4E+07	1.00	9.4E+07
Kr-85m	820	8.8E+07	0.78	6.9E+07
Kr-85	670	8.55E+07	1.00	8.55E+07
Kr-87	3800 1300	1.0E+08 9.6E+07	0.73 0.27	7.3E+07 2.6E+07 9.9E+07
Kr-88	2800 900 520	1.0E+08 9.0E+07 7.6E+07	0.20 0.12 0.68	2.0E+07 1.1E+07 5.2E+07 8.3E+07
Kr-89	4000	1.0E+08	1.00	1.0E+08
Kr-90	2800	1.0E+08	1.00	1.0E+08
Xe-131m	130 160	0.0E+00 1.3E+06	0.58 0.42	0.0E+08 5.5E+05 5.5E+05
Xe-133m	200 230	4.2E+06 7.8E+06	0.62 0.28	2.6E+06 2.2E+06 4.8E+06
Xe-133	350	3.55E+07	1.00	3.55E+07
Xe-135m	500	7.3E+07	0.20	1.5E+07
Xe-135	910 550	9.0E+07 7.8E+07	0.97 0.03	8.73E+07 2.3E+06 8.9E+06
Xe-137	4000 3600	1.0E+08 1.0E+08	0.67 0.33	6.7E+07 3.3E+07 1.0E+08
Xe-138	2400	1.0E+08	1.00	1.0E+08

Example Calculations for Noble Gas Releases

Nuclide	Detection Efficiency (cpm)/(uCi/cc)	Reference Nuclide (cpm)/uCi/cc)	Re_i uCi/cc Xe-133/cpm uCi/cc/cpm
Ar-41	9.4E+07	3.55E+07	2.6
Kr-85m	6.9E+07	3.55E+07	1.9
Kr-85	8.55E+07	3.55E+07	2.4
Kr-87	9.9E+07	3.55E+07	2.8
Kr-88	8.3E+07	3.55E+07	2.3
Kr-89	1.0E+08	3.55E+07	2.8
Kr-90	1.0E+08	3.55E+07	2.8
Xe-131m	5.5E+05	3.55E+07	0.015
Xe-133m	4.8E+06	3.55E+07	0.14
Xe-133	3.55E+07	3.55E+07	1.0
Xe-135m	1.5E+07	3.55E+07	0.42
Xe-135	8.9E+07	3.55E+07	2.5
Xe-137	1.0E+08	3.55E+07	2.8
Xe-138	1.0E+08	3.55E+07	2.8

Table B3-3: Noble Gas Detector, RD-52, Response to Single Nuclide

Nuclide	Limiting Stack Whole Body (uCi/cc)	Concentration Skin (uCi/cc)	Limiting Count Rate (cpm)	Indicated Response (uCi/cc Xe-133)
Ar-41	1.14E-04	4.63E-04	1.1E+04	3.0E-04
Kr-85m	8.59E-04	2.13E-03	5.9E+04	1.7E-03
Kr-85	6.24E-02	4.44E-03	3.8E+05	1.1E-02
Kr-87	1.70E-04	3.64E-04	1.7E+04	4.7E-04
Kr-88	6.84E-05	3.13E-04	5.7E+03	1.6E-04
Kr-89	6.05E-05	2.06E-04	6.1E+03	1.7E-04
Kr-90	6.44E-05	2.38E-04	6.4E+03	1.8E-04
Xe-131m	1.10E-02	9.29E-03	5.1E+03	1.4E-04
Xe-133m	4.00E-03	4.44E-03	1.9E+04	5.6E-04
Xe-133	3.42E-03	8.64E-03	1.2E+05	3.4E-03
Xe-135m	3.22E-04	1.36E-03	4.8E+03	1.4E-04
Xe-135	5.55E-04	1.51E-03	4.9E+04	1.4E-03
Xe-137	7.08E-04	4.35E-04	4.4E+04	1.2E-03
Xe-138	1.14E-04	4.20E-04	1.1E+04	3.2E-04

NOTE: The limiting stack concentrations for whole body and skin listed above were calculated using Equations 3.2c and 3.2d. The limiting count rate and indicated response are calculated using the more restrictive limiting stack concentration as shown below:

Limiting Count Rate= Stack Concentration * E
Indicated Response = Stack Concentration * Re_i

4.0 Off-site Dose Calculations

4.1 Liquid Releases

4.1.1 Control Requirements

Control 3.11.1.2 of Part A of the ODCM requires that cumulative dose contribution estimates be calculated once every 31 days. The cumulative dose contributions should consider the dose or dose commitment to a MEMBER OF THE PUBLIC at or beyond the site boundary from radionuclides in liquid effluent releases. Such releases are limited to ensure that projected doses from each unit are:

a. less than or equal to 1.5 mrems to the total body and less than or equal to 5 mrems to any organ during any calendar quarter, and;

b. less than or equal to 3 mrems to the total body and less than or equal to 10 mrems to any organ during any calendar year.

If the above dose guides are not met, a report must be filed with the NRC Region IV office as required by 10CFR50, Appendix I.

4.1.2 Implementation of Control 3.11.1.2

In order to satisfy the requirements of Control 3.11.1.2, the individuals who suffer the maximum total body and organ doses due to liquid effluent releases are identified. The appropriate total body and organ doses, Dose(a,j), are calculated once a month for fish ingestion and shoreline exposure for each potentially exposed individual (Little Robbins area, Colorado River, and Matagorda Bay/Gulf). These doses are summed for both pathways at each location and compared with the limits of Control 3.11.1.2.

$$\text{Dose}(a,j) = \sum_{\text{path}} \sum_{i} Q(i) * R(a,i,j)_{\text{pathway}} \quad \text{(mrem)} \qquad \text{Eq.4.1a}$$

where Q(i) and R(a,i,j) are described in Table B4-2 and where the values for R(a,i,j) are taken from Table B4-7a. The applicable pathways for doses due to liquid effluents are listed in Table B4-4.

4.2 Liquid Exposure Dose Model

4.2.1 Pathways for Radionuclide Ingestion by Man

Radionuclides which have been released from either unit, mix with the water of the reservoir. These nuclides are expected to be further diluted into the Colorado River with blowdown operations or releases via the spillway overflow (following unusually heavy rains). Water containing trace amounts of radionuclides may diffuse through the bottom of the reservoir and become mixed with shallow ground water. Hydraulic relief wells about the reservoir perimeter may include in their discharge some of this diluted radionuclide-bearing water. These discharges enter the Colorado River, the West Branch Colorado River, and Little Robbins Slough (composed of both branches of Little Robbins Slough; sometimes called West Little Robbins Slough and East Fork Little Robbins Slough). These streams discharge into Matagorda Bay.

4.2.1.1 Colorado River Environment The Colorado River is used primarily for sport fishing and occasionally for barge traffic. No municipal water supplies lie downstream from the plant discharge structure and none are likely to be developed because of the high salt content of the river in this area. A few water use permits allow irrigation of crop land with water taken downstream from the plant, but these permits are seldom (if ever) exercised.

STP possesses Environmental Protection Agency and Texas Department of Water Resources permits which allow the plant to discharge cooling reservoir water only if the river flow exceeds 800 cfs. The average flow rate of the Colorado is about 600 cfs which means blowdown can only occur in rainy periods when river flow is higher than 800 cfs (about 40% of the time). Because such planned discharges and any unplanned spillway releases are likely to occur only during rainy periods, no irrigation is likely with water bearing plant-released radionuclides even if the other water use permits were active. Therefore, no individual or population dose estimates are made on the basis of irrigation with surface water containing radionuclides originating from STP reservoir releases.

The only credible pathway available for internal exposure is the consumption of sea trout, red drum, flounder, catfish, crabs, and shrimp taken from the river by sports fishermen.

Since two small communities are built on the river, one near the discharge facility (Selkirk Island) and the other about seven miles downstream (Matagorda), external exposure is also possible due to shoreline deposits. A number of recreational cabins and trailers also line the east shore of the river south of Matagorda to the Gulf of Mexico (see Figures B4-1 and B4-2).

4.2.1.2 Little Robbins Slough Environment Little Robbins Slough drains through a marsh accessible to local land owners only. Freshwater fish are taken from ponds in this area for sport. However, the annual take is normally small and limited to a few families. Also, some cattle graze in areas where water from Little Robbins Slough might be ingested; however, water for cattle in the area is typically supplied by wells rather than surface water. Hence, no meat ingestion pathways are considered for liquid effluents.

4.2.1.3 Matagorda Bay and the Gulf of Mexico The Colorado River, West Branch Colorado, Little Robbins Slough, and the East Fork Little Robbins Slough all discharge into Matagorda Bay which connects to the Gulf of Mexico as shown in Figure B4-1. Because these bodies of water are connected by natural and man-made channels and the resulting circulation patterns are unknown, no mixing models are available to predict concentrations. However, the average flows of these discharges into Matagorda Bay are small compared with the volume of Matagorda Bay moved to the Gulf of Mexico by tide action. The Matagorda Bay concentration determines the doses due to saltwater pathways and may be assumed to be determined by the ratio of the activity reaching the bay each day and the volume of water moved by tide action (193,820 acre-ft/day).

Internal dose from nuclides reaching Matagorda Bay or the Gulf of Mexico is due to the consumption of sea trout, red drum, and flounder by sports fishermen, and crabs, shrimp, and oysters taken both commercially and by sportsmen.

Since the town of Palacios is built on the shores of Tres Palacios Bay which mixes with West Matagorda Bay, external exposure due to shoreline deposits is possible.

4.2.2 Model for Reservoir Related Radionuclide Decay and Release Off-site

A generally conservative calculation of the off-site dose is accomplished using off-site liquid effluent releases estimated according to the method described in this section.

Table B4-1 lists fractions as calculated by this method for each radionuclide anticipated to be discharged from the plant to the reservoir. These fractions represent the portion of a particular liquid effluent discharge from the plant which will eventually leave the site. These fractions are different for each release route from the reservoir and consist of the product of the variable "Floss" and one or more of the variables "fc, fwc, flrs, and felrs" as described below.

4.2.2.1 Model of the Annual Average Liquid Off-site Release Estimates Based on Plant Discharges to the Reservoir

Radioactive materials released from STP into the main cooling reservoir do not expose members of the public because the reservoir use is restricted. The water is not used for irrigation or drinking and fishing is controlled to prevent ingestion by members of the public. However, a fraction of radioactive material released into the reservoir may eventually leave the reservoir from blowdown activities, overflow, or seepage. The variable "Floss" developed in this section represents the fraction of the activity for a given nuclide which may eventually escape the reservoir through these three mechanisms. The mathematical derivation of the Floss variable follows.

Assumptions:

1. Activity released to the reservoir is not available for release off-site for two weeks, during which time it becomes mixed with previous releases. The mass flow of the reservoir water is such that it should take about two weeks for water to work its way around to the spillway. After one complete circuit of the reservoir (about three weeks), a given release should have mixed into a much larger volume of water than was the original batch release.

2. Batch releases of liquid effluents to the reservoir are made every day or two and are about the same magnitude. Consequently, they approximate a constant discharge rate (Ci/yr). This assumption along with the travel time of assumption #1 above helps assure that the radionuclides in the reservoir are fairly uniformly mixed.

3. The releases due to seepage and blowdown are constant and continuous (any release over the spillway is small and considered to be part of the routine blowdown activity). This assumption is accurate for the seepage, but is only accurate for blowdown if large averaging times are considered. The model is based on annual averages which helps to smooth the discrete blowdown operations each year to approximate a continuous activity.

4. The rate that radioactivity is lost from the reservoir is proportional to the amount of activity in the reservoir at any time. This assumption allows all losses from the reservoir to be treated mathematically the same way as radioactive decay. This assumption is accurate insofar as long averaging times allow discrete discharges to the reservoir and discrete releases from the reservoir off-site to approximate continuous processes.

5. Evaporation from the reservoir offers a release method for tritium and noble gases, but does not affect any other radionuclides. Hence, the release rate constant for tritium will be different than for non-volatile radionuclides.

6. The volume of the reservoir remains constant. A steady state assumption to simplify the model.

7. Five (5) percent of the radioactive material (100% of tritium) discharged from the plant to the reservoir remains in solution and available for release from the reservoir to the off-site environment per EPRI STPEGS MCR Bottom Sediment Characterization Study, 1991, by Richard E. Lockwood (STP) and David R. Blankinship (Texas A&M University).

Estimation of Remaining Batch Discharge as a Function of Time.

The remaining radioactivity, A(t), for a given radionuclide as a function of time after a single discharge of plant effluent mixes into the reservoir is related to the fully mixed discharge activity, A_o, as described below:

$$A(t) = A_o * e^{[-(Y+Yr)*t]} \qquad \text{Eq.4.2a}$$

where:

Y	=	release rate constant for water from the reservoir, per day;
Yr	=	the radioactive decay rate for the given nuclide, per day;
(Y+Yr)	=	total loss rate (release rate and radioactive decay rate) from the reservoir, per day;
t	=	time since mixing in reservoir is complete (14 days after discharge) in days;
A_o	=	activity available for release from the reservoir following a discharge of activity, A_i, from the plant to the reservoir including a mixing delay of 14 days, Ci;
A(t)	=	current activity following mixing of a radionuclide from a plant discharge to the reservoir, Ci.

Release Rate From the Reservoir

The rate of release for a given nuclide from the reservoir is a function of time since discharge from the plant to the reservoir as shown below:

release rate = (activity in the reservoir) * (release rate constant)

since Y = release rate constant (per day)
an A(t) = amount of activity in the reservoir at time "t"

then release rate = A(t) * Y

and substituting for A(t) from Equation Eq. 4.2a

$$\text{release rate} = Y * A_o * e^{[-(Y+Yr)*t]} \qquad \text{Eq. 4.2b}$$

Integrated Release From the Reservoir

The total release during any period of time can be estimated by integrating the release rate of Equation Eq. 4.2b above and evaluating it for that time period.

$$\text{Total release} = \int_{t=T_i}^{T_f} (\text{release rate})\,dt$$

$$= \int_{t=T_i}^{T_f} A_o * Y * e^{[-(Y+Yr)*t]}\,dt$$

$$= \frac{A_o * Y *}{-(Y+Yr)} \int_{t=T_i}^{T_f} e^{[-(Y+Yr)*t]}\,dt$$

$$= \frac{A_o * Y}{-(Y+Yr)} * \left(e^{[-(Y+Yr)*Tf]} - e^{[-(Y+Yr)*Ti]}\right)$$

$$= \frac{A_o * Y}{(Y+Yr)} * \left(e^{[-(Y+Yr)*Ti]} - e^{[-(Y+Yr)*Tf]}\right) \qquad \text{Eq.4.2c}$$

Example Release Calculation

Examples of how one would expect activity to leave STP following a discharge to the reservoir from the plant follow. Three radionuclides are illustrated: a long-lived nuclide such as Cs-137; a nuclide of moderate half-life such as Co-60; and a short-lived nuclide such as Fe-59.

Value of integral from year "T_i" to year "T_f" using Equation 4.2c with three values of Yr.

T_i	T_f	Yr=6.3E-5 per day	Yr=3.4E-4 per day	Yr=2.3E-2 per day
0	1	0.0024 A_o	0.0023 A_o	2.92E-04 A_o
1	2	0.0023 A_o	0.0020 A_o	0.000 A_o
2	3	0.0023 A_o	0.0018 A_o	0.00
3	4	0.0022 A_o	0.0016 A_o	0.00
...	...	...	...	...
...	...	...	...	...
...	...	...	...	...
19	20	0.0015 A_o	0.0000 A_o	0.00
...	...	...	...	...
...	...	...	...	...
...	...	...	...	...
		0.0000 A_o	0.0000 A_o	0.000 A_o
	Total =	0.0950 A_o	0.0190 A_o	2.92E-04 A_o

Discussion

Note from the table above that the release (and hence the off-site dose) following a plant discharge to the reservoir is spread out in time, particularly for the longer-lived nuclides. If we assume that all of a given nuclide which is destined to leave STP does so in the first year, we would assign the dose associated with the release indicated in the last line of the table in the first year and omit the releases listed for subsequent years.

This method is generally conservative since for nuclides with half-lives greater than a couple of years, the dose estimate corresponding to the integrated release is several times larger than the true dose corresponding to the actual release in the first year. The only instance where the method might not be conservative is if in a given year a long-lived nuclide accounted for a large fraction of the 3-mrem limit. If in the following year a short-lived nuclide accounted for the dose, the dose estimate in that second year might be only about 90% of the dose actually delivered that year. This is because the long-lived nuclide from the previous year would still be delivering off-site dose the second year even though the model assigned all that dose the first year. In turn, the short-lived nuclide would deliver virtually all its off-site dose in the year it was actually released to the reservoir.

Conclusion

Considering the uncertainties in estimating off-site flow rates, the possibility of making a 10% error in the off-site doses in consecutive years seems unimportant. Therefore, the ODCM will assign all dose related to the integrated release from the reservoir for a given discharge into the reservoir in the year of the discharge to the reservoir. This integrated release is simply

$$\text{total release} = \frac{A_o * Y}{(Y + Yr)} * \left(e^{[-(Y+Yr)*Ti]} - e^{[-(Y+Yr)*Ti]}\right)$$

$$\text{evaluated with} = \begin{array}{l} Tf = \text{infinity (years) and} \\ Ti = 0 \text{ (years).} \end{array}$$

$$\text{total release} = A_o * \frac{Y}{Y + Yr} \qquad \text{Eq.4.2d}$$

This total release from the reservoir assumes that "Ti" above is measured from the time a radionuclide becomes available for release from the reservoir. Since 14 days must elapse before liquid effluents mix throughout the reservoir, a radioactive decay term, EXP[-Yr*14], should be applied to be strictly correct mathematically. An additional correction factor may be added to account for permanent radionuclide deposition in the reservoir bottom sediments. Five (5) percent of the radioactive material (100% of tritium) released to the reservoir remains in solution. Hence the fraction, Floss, from a given initial plant discharge into the reservoir, A_i, which eventually leaves the reservoir to the uncontrolled off-site environment is

$$\text{Floss} = \frac{\text{total release from site}}{\text{initial release to reservoir}} = \frac{A_o * Y / (Y + Yr)}{A_i} \qquad \text{Eq.4.2e}$$

where $A_o = A_i * \text{EXP}[-Yr*14] * 0.05$ following 14 days of decay and 95% sedimentation.

The fractional loss, Floss, value can be calculated by substituting for the variable A_o in Equation 4.2e.

$$\text{Floss} = \frac{Y / (Y + Yr) * A_i * \text{EXP}[-Yr*14]}{A_i} = \frac{Y}{Y + Yr} * \text{EXP}[-Yr*14]*0.05 \qquad \text{Eq.4.2f}$$

Equation 4.2f is used in section B4.2.2.2 to estimate the fraction of an initial plant discharge into the reservoir which eventually leaves the reservoir to the off-site environment.

4.2.2.2 Liquid Off-site Effluent Release Estimates for Nonvolatile Radionuclides (Evaporation of Tritium and Water Omitted)

The fractions of nuclide "i" from a plant discharge to the reservoir which may eventually reach the off-site environment, $N_c(i)$, $N_m(i)$, $N_{lr}(i)$, are calculated for the three bodies of water into which nuclides might concentrate as below:

Colorado River:	$N_c(i)$	= fc * Floss
Matagorda Bay:	$N_m(i)$	= (fc + fwc + flrs + felrs) * Floss
Little Robbins Slough:	$N_{lr}(i)$	= (flrs + felrs) * Floss

where

Floss = fraction of activity which eventually leaves STP following release to the reservoir from Equation 4.2f

$$= \frac{Y}{Y + Yr_i} * EXP[-Yr_i * 14] * 0.05$$

Y = loss rate due to blowdown and seepage from the nominal reservoir volume
= (annual blowdown flow rate + seepage)/reservoir volume)
= (3400 AF/y + 5700 AF/y) per 150,000 AF = 6.067E-2 per year = 1.662E-4 per day

Yr_i = loss rate due to radioactive decay
= 0.693/(nuclide half-life in days)

fc = fraction of water loss reaching the Colorado River (blowdown plus relief well flow)
= (1027 AF/y + 3400 AF/y) per 9100 AF/y = 0.486

fwc = fraction of water loss reaching the W. Branch Colorado (relief well flow)
= 174 AF/y per 9100 AF/y = 0.019

flrs = fraction of water loss reaching the Little Robbins Slough (relief well flow)
= 2210 AF/y per 9100 AF/y = 0.243

felrs = fraction of water loss reaching the E. Fork of Little Robbins Slough (relief well flow)
= 494 AF/y per 9100 AF/y = 0.054

Reservoir Volume and Flow Data

1. The reservoir volume is fixed at 150,000 AF (nominal volume).
2. The seepage rate is 5700 AF/y to the shallow aquifer (approximately 1800 AF/y remain in the shallow aquifer).
3. The evaporation rate is 38,592 AF/y.
4. The blowdown rate is 3400 AF/y to the Colorado River (anticipated maximum value).
5. Relief well flow to the Colorado River is 1027 AF/y (best estimate).
6. Relief well flow to the W. Branch Colorado River is 174 AF/y (best estimate).
7. Relief well flow to the Little Robbins Slough is 2210 AF/y (best estimate).
8. Relief well flow to E. Fork Little Robbins Slough is 494 AF/y (best estimate).

For example, the fraction of Co-60 reaching the Colorado River, N_c(Co-60), that appears in Table B4-1 is calculated as follows:

Floss	=	Y / (Y + Yr$_i$) * EXP[-Yr*14] *0.05
	=	1.662E-4 / (1.662E-4 + 0.693/1.93E3) * EXP[-0.693/1.93E3*14] * 0.05
	=	0.016
N_c(Co-60)	=	fc * Floss = 0.486 * 0.016 = 7.64E-3

These values of N(i) are used in the equations of sections B4.2.3 to calculate dose to Members of the Public off-site.

4.2.2.3 Tritium Off-site Releases in Liquid Effluents (Evaporative Losses Included)

The fractions of ^{3}H from a plant discharge to the reservoir which may eventually reach the off-site environment must be calculated differently than for the non-volatile nuclides described in section B4.2.2.2. The values of Floss, fc, fwc, flrs, and felrs all have different values because evaporative loses contribute to the reduction of ^{3}H in the reservoir before it can migrate off-site.

$N_c(^3H)$, $N_m(^3H)$, $N_{lr}(^3H)$ = calculated as previously described in section B4.2.2.2

Floss	=	8.712E-04/(8.712E-04 + 1.54E-04) = 0.8498
Y	=	47,690 AF/y per 150,000 AF = 0.3180 per year
	=	8.712E-04 per day
Yr_{H3}	=	0.693/(4506 days) = 1.54E-04 per day
fc	=	(1027 AF/y + 3400 AF/y) per 47,690 AF/y = 9.283E-2
fwc	=	174 AF/y per 47,690 AF/y = 3.649E-3
flrs	=	2210 AF/y per 47,690 AF/y = 4.634E-2
felrs	=	494 AF/y per 47,690 AF/y = 1.036E-2

4.2.3 Off-site Doses from Liquid Effluents

Liquid pathway doses are calculated using the total integrated nuclide releases from the reservoir to the off-site environment. These releases are diluted into the annual average flow of the receiving body of water. Resulting doses will generally overestimate the true off-site values since the activity would normally leave STP over several years and hence would be diluted by substantially more than one year's flow volume once off-site. For example, 50% of the activity contained in the reservoir water is released approximately every 11 years (evaporation excluded); hence, no more than 5.9% of a very long-lived nuclide would leave the site via liquid pathways in any one year. Nevertheless, the projected dose for each release is estimated based upon the assumption that all the activity destined to leave the reservoir does so in the current year. These doses are summed to calculate the month's contribution to the committed dose to the MEMBER OF THE PUBLIC at or beyond the site boundary receiving the greatest dose due to liquid releases. This individual's dose is determined by the consumption of fish and marine invertebrates plus shoreline exposure along the Colorado River, Matagorda Bay or the Little Robbins Slough as calculated below.

4.2.3.1 Fish Ingestion Pathway The pathway dose factors for an individual who ingests saltwater fish, crabs, and shrimp from the Colorado River, Matagorda Bay, or freshwater fish from the Little Robbins

Slough area are calculated using Equation 4.2g where the parameter descriptions are in Table B4-2 and the parameter values are as listed in Table B4-3. The resulting pathway dose factors are tabulated in Table B4-7a by organ and age. The dose commitment age as described in Regulatory Guide 1.109 was used in the tabulation of dose factors in Table B4-7a.

$$R(a,i,j)_{pathway} = 1000 * \frac{U}{M*F} * \sum_{i} N(i)*B(i)*D(a,i,j)*EXP[-Y(i)*T] \quad \text{(mrem/Ci)} \qquad \text{Eq. 4.2g}$$

Equation 4.2g is equivalent to Regulatory Guide 1.109 Equation A-3 methods. It is restated as used in the computer program used to calculate the pathway dose factors at STP and includes a factor, N(i), to account for radioactive decay and sedimentation before leaving the reservoir.

4.2.3.2 <u>Shoreline Deposition Pathway</u> Individuals who live in the area could be exposed to accumulations of contaminated silt deposited along the Colorado River bank, along Little Robbins Slough, or on the shores of Matagorda Bay. The pathway dose factors from these potential shoreline deposits are calculated using Equation 4.2h with the parameters described in Table B4-2 and with values as listed in Table B4-3. The resulting pathway dose factors are compiled in Table B4-7a by organ and age. The dose commitment age as described in Regulatory Guide 1.109 was used in the tabulation of dose factors in Table B4-7a.

$$R(a,i,j)_{shoreexposure} = 110{,}000 * \frac{Ub*W}{M*F} * \sum_{i} N(i)*T(i)*D(a,i,j) * EXP[-Y(i)*T] * (1-EXP[-Y(i)*Tb]) \quad \text{(mrem/Ci)} \qquad \text{Eq. 4.2h}$$

Equation 4.2h is equivalent to Regulatory Guide 1.109 Equation A-7 methods. It is restated as used in the computer program used to calculate the pathway dose factors at STP and includes a factor, N(i), to account for radioactive decay and sedimentation before leaving the reservoir.

4.3 Gaseous Releases

4.3.1 <u>Control Requirements</u>

Control 3.11.2.1 of Part A of the ODCM requires that the dose rate at or beyond the site boundary due to radioactive materials released in gaseous effluents from the site be limited to the following values:

a. The dose rate limit for noble gases must be less than 500 mrem/yr to the total body and less than 3000 mrem/yr to the skin, and

b. The dose rate limit for all radionuclides other than noble gases with half-lives greater than 8 days be less than 1500 mrem/yr to any organ.

These requirements stem from the NRC desire for nuclear power plants to operate at a small fraction of the radiological protection limits of 10CFR20.

Control 3.11.2.2 of Part A of the ODCM also requires that the air dose in areas at or beyond the site boundary due to noble gases released in gaseous effluents shall be limited to the following:

a. During any calendar quarter to less than or equal to 5 mrads for gamma radiation and 10 mrads for beta radiation, and

b. During any calendar year to less than or equal to 10 mrads for gamma radiation and 20 mrads for beta radiation.

Control 3.11.2.3 further limits the dose to a MEMBER OF THE PUBLIC from I-131, I-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas at or beyond the site boundary as follows:

a. During any calendar quarter to less than or equal to 7.5 mrems to any organ, and

b. During any calendar year to less than or equal to 15 mrems to any organ.

These last two requirements stem from STP's commitment to operate STP within the guidelines described in 10CFR50, Appendix I, for maintaining doses to the public as low as reasonably achievable.

4.3.2 Implementation of Control 3.11.2.1

4.3.2.1 Noble Gases All gaseous effluent releases from STP are assumed to be ground level due to the proximity of each unit's vent to the roof. For the purpose of demonstrating that off-site dose rates have not exceeded the dose rate limits of this Control, the atmospheric dispersion factor, X/Q, may be assumed to be 5.3E-06 sec/cubic meter. This represents the 500 hour average X/Q at the site boundary and occurs in the NNW sector. When possible, actual hourly X/Q values coupled with hourly release data are used in place of composite release data and historical average X/Qs.

The hourly average dose rates to the whole body and to the skin due to noble gas releases may be estimated using Equations 4.4d and 4.4e of this section provided the shielding factor, Sf, equals 1.0 for the purpose of determining compliance with Control 3.11.2.1.

4.3.2.2 Iodine and Particulates The maximum dose rate to an organ, j, in a given age group, a, due to particulate releases may be estimated as follows:

$$\text{Dose rate }(a,j)=X/Q^*\sum_i R(a,i,j)_{\text{inhalation}}{}^*Q_i+D/Q^*\sum_i\ \sum_{path} R(a,i,j)_{\text{pathway}})^*Q_i(\text{mrem/hr})\quad \text{Eq. 4.3a}$$

where

Q_i	=	release rate of nuclide "i" (Ci/hr),
X/Q	=	5.3E-06 (sec/m^3) (or actual estimate of X/Q for H3 and C14 or depleted X/Q for particulates and iodines at the time of release),
D/Q	=	8.4E-09 (1 /m^2) (or actual estimate of D/Q at the exposure location),
$R(a,i,j)_{\text{pathway}}$	=	pathway dose factors from Table B4-7b (units as described in notes to Table B4-7).

The highest organ dose so calculated may be used for demonstrating compliance with Control 3.11.2.1. However, only pathways confirmed by the land use census need be considered (e.g. cow-milk-infant pathway need not be considered in the absence of the cow).

4.3.3 Implementation of Control 3.11.2.2

NUREG-0133 allows STP to use the highest calculated annual average X/Q in calculating doses for comparison with the quarterly and annual dose limits. However, NUREG-0133 recommends the use of the highest 500-hour average X/Q for doses due to short-term releases. STP normally has available hourly average X/Q values for each sector plus time-dated release information. When possible, these hourly X/Q values coupled with hourly release data are used in place of composite release data and historical average X/Qs.

The historical dispersion values which may be used for calculations in place of historical averages are:

annual average releases	=	1.1E-06 (seconds per cubic meter)
500 hour or shorter releases	=	5.3E-06 (seconds per cubic meter)

4.3.3.1 Noble Gases The noble gas releases averaged over a calendar quarter or a calendar year result in a dose to air at the site boundary as calculated using Equations 4.4f for gamma radiation and Equation 4.4h for beta radiation.

4.3.4 Implementation of Control 3.11.2.3

4.3.4.1 Iodines, Tritium, and Particulates The dose to a MEMBER OF THE PUBLIC stationed at or beyond the site boundary (Table B4-6) due to radioiodine and particulate releases is estimated using Equation 4.4k and the appropriate pathway dose factor from Table B4-7b. The historical dispersion values (X/Q and depleted X/Q) may be used in place of actual data if necessary as described in part 4.3.3 above.

4.4 Gaseous Dose Models and Dose Formulas

4.4.1 Dispersion Calculation Methods

If current meteorological data are used to estimate dispersion, X/Q, in place of the historical values, calculations for routine releases use the sector-average version of the equations for atmospheric relative concentration. These calculations are made in accordance with the methodology in NRC Regulatory Guide 1.111 and are all based on ground level releases.

4.4.1.1 X/Q Calculation The sector average X/Q for a given hour is calculated using:

$$X/Q = \frac{2.03}{Umn * Dxqc * Smn} \ (sec/m^3) \qquad \text{Eq. 4.4a}$$

where

$Smn = [sz^2 + (Hcon^2/2 * \pi)]^{1/2}$

or

$Smn = sz * (3)^{1/2}$; whichever is less;

and

Hcon = building height (meters),
sz = vertical dispersion coefficient (meters),
Smn = dispersion coefficient with building wake factor included (meters)
Dxqc = downwind distance to the receptor (meters),
Umn = hourly average wind speed (meters/second),
2.03 = $(2/\pi)^{1/2}$ divided by the sector width in radians, $(2*\pi/16)$.
π = 3.14

4.4.1.2 Depleted X/Q Calculation X/Q values are used in conjunction with tritium and noble gases released. However, the downwind concentrations for particulates and radioiodines will be affected by ground deposition. X/Q values used for calculating inhalation doses from particulates and radioiodines must be modified by the ground depletion factors of Table B4-4 (from Figure 2 of NRC Regulatory Guide 1.111).

$$(X/Q)_{depl} = (X/Q) * (\text{ground depletion factor}) \quad (sec/m^3) \qquad \text{Eq. 4.4b}$$

4.4.1.3 Ground Deposition Ground deposition is calculated using the deposition factors of Table B4-5 (also from Regulatory Guide 1.111, Figures 6-9).

$$(D/Q) = \frac{(\text{deposition factor})}{Dxqc * 0.3927} \quad (1/m^2) \qquad \text{Eq. 4.4c}$$

Where 0.3927 = radians in one sector or $(2\pi)/16$,
Dxqc = down wind distance (meters).

Deposition calculated by multiplying this term, D/Q, by the release rate, Q, will yield values independent of atmospheric stability as indicated in NRC Regulatory Guide 1.111.

4.4.2 Submersion Dose From Noble Gases

The methods used to estimate doses due to noble gases are those of Regulatory Guide 1.109. The whole body and skin doses from submersion in a cloud of noble gas may be calculated by multiplying the appropriate dose factor for the plume pathway from Table B4-7b by the dispersion, X/Q, and by the release rate, Q. An equivalent calculation can be accomplished using the formulas described in the following three subsections:

4.4.2.1 Whole Body Dose Rate

$$Dr_{gamma} = 0.114 * X/Q * \sum_i (Q(i) * Dfi_{gamma}) * S_f \text{ (rem/hr)} \qquad \text{Eq. 4.4d}$$

where

0.114	=	conversion factor from (mrem-m^3)/(pCi-yr) to (rem-m^3)/(uCi-hr)
X/Q	=	from Equation 4.4a (sec/m^3)
Q(i)	=	isotope "i" release rate (uCi/sec) from monitor RT-8010B
Dfi_{gamma}	=	gamma dose to tissue conversion factor for nuclide gamma "i" from Table B-1 of Regulatory Guide 1.109 (mrem-m^3/pCi-yr)
S_f	=	0.7, shielding factor from Regulatory Guide 1.109
	=	1.0 when determining compliance with Control 3.11.2.1

Dr_{gamma} of Equation 4.4d is equivalent to $D^T_\infty(r,\theta)$ of Equation B-8 in Regulatory Guide 1.109. Equation 4.4d is expressed as rem/hr whereas Equation B-8 is expressed in units of rem/yr. Equation 4.4d contains factors which exist in the Regulatory Guide as the combination of Equations B-8 and C-3.

4.4.2.2 Skin Dose Rate from Noble Gases Skin dose rate is calculated based on both the beta emissions and gammas coming from the noble gas cloud surrounding the receptor.

$$Dr_{skin} = 1.11 * S_f * Dr_{gamma(air)} + Dr_{beta(skin)} \text{ (rem/h)} \qquad \text{Eq. 4.4e}$$

where
$$Dr_{gamma(air)} = 0.114 * X/Q * \sum_i Q(i) * Dfi_{gamma(air)} \text{ (rad/h)} \qquad \text{Eq. 4.4f}$$

and
$$Dr_{beta(skin)} = 0.114 * X/Q * \sum_i Q(i) * Dfi_{beta(skin)} \text{ (rem/h)} \qquad \text{Eq. 4.4g}$$

S_f	=	0.7, default shielding factor from Regulatory Guide 1.109, Table E-15
	=	1.0 when determining compliance with Control 3.11.2.1
$Dfi_{beta(skin)}$	=	beta dose to tissue conversion factor from Table B-1, Regulatory Guide 1.109 (mrem-m^3/pCi-yr),
$Dfi_{gamma(air)}$	=	gamma dose to air conversion factor from Table B-1, Regulatory Guide 1.109 (mrad-m^3/pCi-yr),
1.11	=	ratio of the mass stopping powers for electrons in air to tissue from Regulatory Guide 1.109 Equation B-9.
Q(i)	=	isotope "i" release rate (uCi/sec) from monitor RT-8010B

Equation 4.4e is equivalent to Equation B-9 in combination with Equation C-3 of Regulatory Guide 1.109. Equations 4.4f and 4.4g were extracted from Equation 4.4e to simplify its expression. The conversion constant was adjusted to provide rem per hour rather than rem per year as found in the Regulatory Guide. Equation 4.4f is also equivalent to Equation B-5 combined with Equation B-4 of Regulatory Guide 1.109.

The gamma dose rate to air is calculated here as an intermediate step in calculating the total dose rate to skin from noble gases. However, this gamma dose rate to air value, Dr gamma(air) from Equation 4.4f is used to demonstrate compliance with the first part of Control 3.11.2.2 if multiplied by the release duration in hours as described in Section B4.4.2.3.

4.4.2.3 <u>Dose to Air from Noble Gases</u> The dose to air at the site boundary is a required dose calculation in Control 3.11.2.2. The first step is to calculate the beta dose rate to air for noble gases as indicated below:

$$Dr_{beta(air)} = 0.114 * X/Q * \sum_i Q(i) * Dfi_{beta(air)} \quad \text{(rad/h)} \qquad \text{Eq. 4.4h}$$

where $Dfi_{beta(air)}$ = beta dose to air conversion factor from Table B-1, Regulatory Guide 1.109 (mrad-m^3/pCi-yr),

0.114 = conversion factor from (mrad-m^3 /pCi-yr) to (rad-m^3/uCi-hr),

X/Q = from Equation 4.4a (sec/m^3),

Q(i) = isotope "i" release rate (uCi/sec) from monitors RT-8010B.

Equation 4.4h contains the elements of Equations B-4 and B-5 of Regulatory Guide 1.109. The dose rates of Equations 4.4f and 4.4h may be multiplied by the release duration to give the dose to air from gamma and beta radiation as shown below:

$$Dose_{gamma(air)} = Dr_{gamma(air)} * T \qquad \text{Eq. 4.4i}$$

$$Dose_{beta(air)} = Dr_{beta(air)} * T \qquad \text{Eq. 4.4j}$$

where $Dr_{gamma(air)}$ is from Equation 4.4f

$Dr_{beta(air)}$ is from Equation 4.4h

T is the release duration

4.4.3 <u>Dose Due to Airborne Radionuclides</u>

The dose delivered to the individual with the highest exposure due to airborne radioactive particles and gases is calculated as the sum of pathway doses for all nuclides present.

$$Dose_{air}(a, j) = (\text{dispersion}) * \sum_{path} \sum_{i} Q(i) * R(a, i, j) \quad \text{(mrem)} \qquad \text{Eq. 4.4k}$$

where

dispersion = ground deposition, D/Q (1/m^2), for ingestion and deposition pathways, or
$(X/Q)_{depl}$ (sec/m^3), for particle inhalation pathways, or
(X/Q) (sec/m^3), for noble gas, H-3, and C-14 all pathways

Q(i)	=	integrated release of nuclide "i" stored by plant computer from monitors RT-8010B (Ci),
R(a,i,j)	=	age, nuclide, and organ specific dose factor for a given pathway as listed in Table B4-7b (units as described in notes to Table B4-7).

For ingestion pathways involving particles, the ground deposition as calculated from Equation 4.4c is used for the dispersion in Equation 4.4k.

For inhalation of particles, the depleted X/Q from Equation 4.4b is used for the dispersion in Equation 4.4k.

For both ingestion and inhalation of H-3 and C-14, the X/Q from Equation 4.4a is substituted for dispersion in Equation 4.4k.

For plume immersion dose to noble gases, the X/Q from Equation 4.4a is substituted for dispersion in Equation 4.4k.

Although in practice these calculations are performed at the distances and directions listed in Table B4-6, only the distance and direction giving the largest organ dose is used in Equation 4.4k above.

The exposure pathway dependent dose factors, R(a,i,j), of Table B4-7b were generated using a code similar to NRC's GASPAR routine as described in NUREG-0597. The pathways for radionuclides released to the atmosphere which may expose the local population do not include the milk pathway. No milk cows or goats have been identified within five miles of the plant, and no commercial dairies exist within fifty miles of the plant. Since a milk cow or goat could be introduced in the future, Table B4-7b contains dose factors for those pathways even though they are not used at this time. These dose factors of Table B4-7b were calculated for the pathways, organs, and age groups as described below:

Pathways	*Pathway Description*
Plume Immersion	Whole body and skin exposure to noble gas
Ground	Whole body and skin exposure to particulates deposited on ground
Vegetation Ingestion	Organ doses to particles, ^{3}H, and ^{14}C deposited on vegetation
Meat Ingestion	Organ doses to particles, ^{3}H and ^{14}C in meat products
Cow Milk	Not currently an active pathway
Goat Milk	Not currently an active pathway
Inhalation	Organ doses to inhaled particles, ^{3}H, and ^{14}C

Age Group	*Years of age (yr)*	*Dose Commitment Age (yr)*	*Fraction of population in each age group*
Infant	0 - 1	0	0.0
Child	1 - 11	4	0.18
Teen	11 - 17	14	0.11
Adult	17 ->	17	0.71

Organs
Total Body
G.I. Tract
Bone
Liver
Kidney
Thyroid

Lung
Skin

The version of GASPAR used to calculate the dose factors for ingestion employed the methods of Regulatory Guide 1.109 Equation C-13. Default parameter values as contained in the Regulatory Guide were used in conjunction with Equation C-13 and its supporting equations, except for ^3H where the site specific humidity value was set to 13 g/m^3. Equation C-13 contains concentration expressions for meat, milk, produce, and leafy vegetables. Each pathway dose factor of Table B4-7b was calculated by setting the other pathway concentration expressions to zero and solving the equation for each organ and age group. The airborne concentration ($Q(T) * \chi/Q$) or ground deposition ($d_i(r,\theta)$) was extracted from the components of Equation C-13 (Equations C-5, C-8, C-9, C-10, C-11, and C-12) to render the pathway dose factors of Table B4-7b independent of the release rate and atmospheric diffusion/deposition. As required by the Regulatory Guide, ^{14}C and ^3H concentrations in vegetation were calculated using special equations (C-8 and C-9).

The inhalation dose factors of Table B4-7b were calculated by GASPAR using the methods of Equation C-4 with the airborne concentration factor of Equation C-3 extracted to render dose factors independent of release rate and atmospheric dispersion.

The ground shine dose factors of Table B4-7b were calculated by GASPAR using the methods of Equation C-2 with the deposition factor ($\delta_i(r,\theta) * Q(i)$) of Equation C-1 extracted to render dose factors independent of release rate and location.

The dose commitment age as described in Regulatory Guide 1.109 was used in the tabulation of dose factors in Table B4-7b.

4.5 Control 3.11.1.3

The liquid waste processing system shall be operable and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from each unit, to unrestricted areas would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31-day period.

Doses due to liquid effluent releases shall be estimated prior to release of each batch from the radioactive waste monitor tanks. The 31-day dose projection shall be calculated as shown below:

$$\text{31-day dose projection} = \frac{31}{\text{days}} * (\text{accumulated dose}) + \text{SF} \quad (\text{mrem}) \qquad \text{Eq. 4.5a}$$

where

31 = days in the averaging period

days = integer number of days into the quarter (greater than or equal to 1 but less than or equal to 92)

accumulated dose = sum of doses from releases in the current quarter (mrem) and the projected release

SF = safety factor, (projected dose limit) * 0.05, i.e. 5% of limit

Since this operating condition is applied to each unit separately, concurrent releases from both units need not be considered.

4.6 Control 3.11.2.4

The gaseous waste processing system shall be operable and appropriate portions of this system shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases from each unit to areas at or beyond the site boundary would exceed:

0.2 mrad to air from gamma radiation, or
0.4 mrad to air from beta radiation, or
0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

Unit vent air samples are analyzed weekly for each unit. The average concentrations of the radionuclides so measured may be used to calculate the unit specific doses from releases that week. These average weekly doses plus doses from any special or batch releases during the week may be used in Eq. 4.5a to project the doses of Control 3.11.2.4 over the subsequent 31-day period. If an unusually large release is planned, add this projected dose to the average 31-day doses to confirm that the operating constraints of Control 3.11.2.4 are satisfied. These constraints pertain to each unit separately, and the dose projections from the two units need not be summed when determining operating constraints imposed by this Control.

4.7 Control 3.11.4 Dose Calculations

If the annual dose or dose commitment to a MEMBER OF THE PUBLIC at or beyond the site boundary due to releases of liquid or gaseous effluents exceeds twice the limits of Controls 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, Control 3.11.4 requires that the total dose from the uranium fuel cycle be calculated.

Since no mining, milling, or waste disposal activities exist within 50 miles of STP, only direct radiation from plant structures need be added to that calculated for effluents to obtain the total dose. Direct radiation from the plant and plant structures is estimated based on ambient radiation measurements made in the proximity of each potential source within a direct line of sight to the critical receptor location. TLD measurements at the protected area fence may provide the estimate of direct radiation following background subtraction. This measured dose rate may be adjusted to compensate for distance to the critical receptor location. The direct radiation dose shall be added to the doses previously calculated for radioactive effluents for comparison with the limits of 40CFR Part 190 as shown below:

$$\text{Total Dose} = \textbf{MAXIMUM} \{\text{Dose}_{\text{liquid}}(a,j) + \text{Dose}_{\text{air}}(a,j) + \text{Dose}_{\text{direct}}\} \quad \text{from age groups, a, and organs, j}$$

where

$\text{Dose}_{\text{liquid}}(a,j)$		from Equation 4.1a
$\text{Dose}_{\text{air}}(a,j)$		from Equation 4.4k
$\text{Dose}_{\text{direct}}$	=	$\text{TLD} * (\text{PA})^2 / (\text{RD})^2$
TLD	=	net measured exposure @ protected area fence, rem
PA	=	containment to protected area fence distance where TLD located, meters
RD	=	distance from containment to critical receptor, meters

4.8 10CFR20.1301 Dose to MEMBERS OF THE PUBLIC

In addition to meeting the requirements of the controls in Part A of the ODCM and 40CFR190, STP is required to meet the dose limits of 10CFR20.1301 in accordance with the methods of 10CFR20.1302. As provided in 10CFR20.1302(b)(1), the calculated dose to a Member of the Public must not exceed 100 mrem in a year. For the purpose of 10CFR20.1301 dose calculations, a Member of the Public is an individual inside or outside the owner controlled area except when that individual receives an occupational dose. Occupational dose at STP is associated with individuals whose work may involve exposure to radiation in excess of 100 mrem in a year. All such individuals are given one or more levels of training as required by 10CFR19 for occupational exposed workers.

All STP employees and contractors with unescorted access to the protected area (the security fence surrounding the nuclear two units) have been identified as receiving occupational dose and meet the training requirements of 10CFR19. Vendors or working visitors (temporary contractors) who are likely to receive 100 mrem in a year are briefed on the hazards of ionizing radiation, are assigned a dosimeter, and their exposure is controlled to occupational limits.

STP employees and contractors whose dose is not occupational are considered Members of the Public and do not take any radiation worker training courses (General Employee Training Classes). In addition, some visitors who enter the protected area but are not expected to receive occupational exposure are considered Members of the Public. They are not trained or issued a radiation dosimeter. The dose to a Member of the Public as described here is not part of the off-site dose calculated to demonstrate compliance with 40CFR190 or the controls of Part A.

To demonstrate compliance with 10CFR20.1301, dose to Members of the Public from inhalation of particles, iodines, and tritium is added to dose from exposure to external sources and noble gases. No dose is estimated for liquid effluents since no pathway exists for Members of the Public to ingest contaminated water while at STP. The sum of these two components must not exceed 100 mrem in a year.

The annual inhalation and noble gas dose components are calculated as the sum (for both units) of the annual average concentrations of particles, iodines, and noble gases in the highest X/Q sector at 200 meters divided by the corresponding concentrations in Appendix B, Table 2, column 1, and multiplied by 50 mrem/year.

$$\text{inhalation dose} = 50 * T * X/Q_{200} \sum_{\text{nuclide } i} (Q(i) / EC_i) \qquad \text{Equation 4.8a}$$

where

$Q(i)$	=	annual average release rate of nuclide i, Ci/sec
EC_i	=	effluent concentration of nuclide i, uCi/cm^3 in air
X/Q_{200}	=	maximum diffusion constant at 200 meters, sec/m^3
50	=	mrem for exposure to 1 EC for a year, mrem/yr
T	=	exposure duration, yr

The external dose rate component (above natural background) may be measured using TLDs within the owner-controlled area.

An example for a visiting Member of the Public to the protected area follows:

> The visitor makes four entries into the protected area during the year to service equipment. Each visit is assumed to involve a 10 hour work day for a total exposure time of 40 hours over the course of the year. While inside the protected area fence, he is exposed to airborne Xe-133, I-131, and Co-60. The annual average X/Q at 200 meters is 2.4E-5 m^3/sec and the annual average release rates summed for both units are 1.0E-6, 2E-15, and 8E-13 Ci/sec respectively. The maximum external dose rate measured by TLDs is 100 mrem/yr. (Note: The external dose rate alone is limited by 10CFR20.1301 to 2 mrem in an hour.)

$$\begin{aligned}\text{inhalation/immersion dose} &= 50 * 40 \text{ hr} / 8760 \text{ (hr/yr)} * 2.4E\text{-}5 * \sum(1.0E\text{-}6)/(5.0E\text{-}7) \\ &\quad + (2E\text{-}15)/(2E\text{-}10) + (8E\text{-}13)/(5E\text{-}11) \\ &= 1E\text{-}5 \text{ mrem}\end{aligned}$$

external dose	=	100 mrem/yr * 40 hr / 8760 (hr/yr)
	=	0.5 mrem

total dose = 0.5 mrem in a year for a visiting Member of Public within the protected area for comparison with 10CFR20.1301 annual dose limits.

An example of an employee Member of the Public located outside the protected area follows:

The employee works 2000 hours while not exposed to occupational dose during the year. The work location is in an office building near the protected area fence where he is exposed to airborne Xe-133, I-131, and Co-60. The annual average X/Q at 200 meters is 2.4E-5 m^3/sec and the annual average release rates summed for both units are 1.0E-6, 2E-15, and 8E-13 Ci/sec respectively. The maximum external dose rate due to plant operations measured by TLDs at the protected area fence is less than 20 mrem/yr.

inhalation/immersion dose rate	=	50 * 40 hr / 8760 (hr/yr) * 2.4E-5 * $\sum$ (1.0E-6)/(5.0E-7) + (2E-15)/(2E-10) + (8E-13)/(5E-11)
	=	1E-5 mrem
external dose		< 20 mrem/yr * 2000 hr / 8760 (hr/yr)
		< 5 mrem

total dose < 5 mrem in a year for an employee Member of Public located outside the protected area for comparison with 10CFR20.1301 annual dose limits.

4.9 Population Dose Estimation

Doses to the population are calculated in a manner similar to that described for individuals with two exceptions. The dose factors are taken from Table B4-11, and the doses calculated for each population group are summed. The R(all,i,j) age-adjusted dose factors for atmospheric pathways of Table B4-11 were calculated using the equations of Regulatory Guide 1.109 in the GASPAR code along with default consumption/use factors. The values for R(all,i,j) appearing in Table B4-11 for liquid releases are the age adjusted dose factors for the general population calculated as [0.71 * R(adult,i,j) + 0.11 * R(teen,i,j) + 0.18 * R(child,i,j)]. R(a,j,i) are calculated from Eq. 4.2g and Eq. 4.2h using data from Table B4-8.

Population doses due to liquid effluents are calculated in the manner of Equation 4.9a for each member of the population. The resulting doses are then multiplied by the number of individuals residing within 50 miles of STP. If sufficient quantities of a particular food are produced within 50 miles of STP to feed the 300,000 inhabitants of this region, the population for that pathway is reduced to the number who could consume the average amount of that food without exhausting the locally produced supply. For example, since only about 45,000 kg of saltwater sport fish are taken in Matagorda Bay and the Colorado River each year, only 3800 individuals may be assumed to consume 5.9 kg = (0.71 * 6.9 + 0.11 * 5.2 + 0.18 * 2.2) per year of fish each to account for this mass. In order to account for recreation on both the Colorado River and Matagorda Bay, half the population is assumed to use each of these waters. All crustaceans (1.8 E+06 kg) per year are assumed to be taken from Matagorda Bay.

$$\text{pop. dose}_{liq} = \sum_{path=p} \text{population}_p \sum_{nuclide=i} Q(i) * R(all,i) \qquad \text{Eq. 4.9a}$$

where

population_p	=	population within 50 miles exposed to each pathway, P
Q(i)	=	release by nuclide, i (Ci)
R(all,i)	=	are taken from Table B4-11 for the whole body

Population doses due to gaseous effluents are calculated in a two step process. The population within 50 miles of STP is listed by sector and distance in Table B4-10. The population dose is calculated by first calculating the X/Q, depleted X/Q, and D/Q for each distance and sector. The product of the dose factors from Table B4-11 and X/Q (for the plume pathway), depleted X/Q (for the particulate and iodine inhalation pathway), or D/Q (for the ingestion pathways) for a given distance/sector group gives the dose to each member of that group. The product of these doses by the number of individuals in the group gives the dose to each group. The sum over all groups within 50 miles gives the total population dose.

Equation 4.9b:

$$\text{pop. dose}_{air} = T * \sum_{s} pop(s) * [X/Q(s) * \sum_{i} R(all,i)_{plume} * Q(i) +$$
$$X/Q_d(S) * \sum_{i} R(all,i)_{inhalation} * Q(i) +$$
$$D/Q(s) * \sum_{path} \sum_{i} R(all,i)_{path} * Q(i)]$$

where

T	=	time period covered by the calculation (hours)
pop(s)	=	number of people in distance/sector group "s" from Table B4-10
X/Q(s)	=	X/Q for distance/sector "s" per Eq. 4.4a (sec/m^3)
$X/Q_d(s)$	=	depleted X/Q for distance/sector "s" per Eq. 4.4b (sec/m^3)
	Note:	X/Q(s) substituted for $X/Q_d(s)$ for H-3 and C-14
R(all,j)	=	dose factors from Table B4-11 for the whole body for each pathway (units as described in notes to Table B4-7)
Q(i)	=	release rate of nuclide "i" (Ci/sec)
D/Q(s)	=	deposition for distance/sector "s" per Eq. 4.4c (1/m^2)

4.10 Dose Due to Tritium in Liquid Effluents That Become Airborne

Each nuclear plant discharges tritium to the main cooling reservoir. Since the tritium concentration in the reservoir is relatively stable, the annual discharge must about equal the annual release. Although a small fraction of the tritium enters the shallow aquifer and some decays, the majority must exchange with atmospheric moisture and leave the site. The ODCM does not model this release path and assumes as specified in the Regulatory Guides that liquid releases remain in liquid dose pathways. This assumption may omit some offsite dose. In 2005 (see Condition Report 05-8815 Action #6) STPNOC evaluated this pathway and determined that the omitted dose could be a significant fraction of the dose calculated using historical pathway analyses.

In the course of completing Condition Report #05-8815, Action #6, the then approved Environmental Protection Agency code Isclt3 was found to apply to area sources like the main cooling reservoir. 2004 joint frequency table data were used in conjunction with the Isclt3 code to calculate an estimate of the X/Q values for the site boundary and the nearest resident in each of 16-sectors as listed in the ODCM, Revision 12. The meteorological data for 2004 are typical and generate point source X/Q values consistent with those listed in the UFSAR. The area source X/Q values calculated using the Isclt3 code are listed in the following table:

CN-2883

The particulate channel is used as part of the Reactor Coolant Pressure Boundary (RCPB) leakage detection system. The sensitivity and response time of this part of the leakage detection system, which is used for monitoring unidentified leakage to the Containment, are sufficient to detect an increase in leakage rate of the equivalent of one gal/min within one hour. Elements of this monitor, including the indicator mounted in the RMS CR cabinet, are designed and qualified to remain functional following a Safe Shutdown Earthquake (SSE), in compliance with RG 1.45. Further information on the RCPB leakage detection system is presented in Section 5.2.5.

11.5.2.3.3 Unit Vent Monitor: The unit vent monitor samples the plant vent stack prior to discharge to the environment and monitor for particulates, iodine, and noble gases.

The unit vent particulate and iodine monitor draws representative air samples from the plant vent stack via isokinetic nozzles in the stack, and directs them through a moving filter paper monitored by a shielded beta-sensitive scintillation detector. The sample stream then passes through a charcoal collector, where collected iodine is monitored by a shielded gamma-sensitive scintillation detector. The sample is then returned to the vent stack.

A separate wide-range gas monitor is provided for the unit vent. The monitor has two isokinetic nozzles for sampling during both normal and accident conditions. The stack samples pass first through a sample conditioning unit which filters particulates and iodine and may be used to take grab samples. The samples then pass through the shielded detector assembly, which uses three detectors to cover the complete range required. The low range detector uses a beta-sensitive plastic scintillator-photomultiplier (PM) tube. The mid-range and high-range detectors use cadmium telluride (CdTe), chlorine-doped, solid-state sensors. This wide-range gas monitor satisfies the requirements of NUREG-0737, Item II.F.1 for provisions for sampling plant effluents for iodines and particulates and for noble gas effluents from the plant vent.

11.5.2.3.4 Control Room Electrical Auxiliary Building Ventilation Monitors: The CR/EAB ventilation monitors are Class 1E monitors which continuously assess the intake air to the CR for indication of abnormal airborne radioactivity concentration. Each monitor assembly is powered from a separate electrical power source. In the event of high radiation CR emergency ventilation operation is initiated (Section 7.3.2). Failure of a monitor is alarmed in the CR.

Each monitor assembly is comprised of a recirculation pump, beta-sensitive scintillation detector, four-pi lead shielding, check source, stainless steel sample gas receiving chamber, and associated electronics.

11.5.2.3.5 Condenser Vacuum Pump Monitor: Gaseous samples are drawn through an off-line system by a pump from the discharge of the vacuum pump exhaust header of the condenser. This channel monitors the gaseous sample for radioactivity which would be indicative of an SG tube leak, allowing reactor coolant to enter the secondary side fluid; this monitor complements the SGBD monitors in indication of a SG tube leak. The gaseous radioactivity levels are monitored by a single detector in a manner similar to the unit vent wide range gas monitor.

11.5.2.3.6 Spent Fuel Pool Exhaust Monitors: The SFPE monitors are Class 1E and are identical to the CR/EAB ventilation monitors described in Section 11.5.2.3.4 except that they sample the exhaust from the FHB. In the event of high radiation the monitors initiate emergency operation of the FHB HVAC, causing the exhaust air to be filtered prior to release (Section 7.3.3). Failure of a monitor is alarmed in the CR.

11.5.2.3.7 RCB Purge Isolation Monitors: The RCB/purge isolation monitors are Class 1E monitors that sample the Containment Normal Purge System or the Supplementary Purge System and are identical to the CR/EAB ventilation monitors described in Section 11.5.2.3.4. In the event of high radiation the monitors send signals to the Solid-State Protection System (SSPS) for containment ventilation isolation (Section 7.3.1). Failure of a monitor is alarmed in the CR.

11.5.2.4 Liquid Monitors. Fixed, off-line monitors are provided for continuous detection and measurement of radioactivity for liquid process streams. The design parameters for these monitors are summarized in Table 11.5-1. Each monitor is provided with demineralized water for flushing.

11.5.2.4.1 Sampling Devices: For each monitor, a sample is drawn from the process line, passed through a shielded sample chamber, through the sample pump and then returned to the system. Each sample pump is capable of drawing at least one gal/min of liquid through the monitor. The sample flow rate is controlled by means of a manual valve.

Each monitor has a low-sample-flow alarm.

The monitor inlet and outlet lines have compression fittings. The sample piping has isolation valves so that the monitor skid can be isolated and the sample chamber dissembled for decontamination.

11.5.2.4.2 Detector Unit: Each detector is a NaI(Tl) gamma-sensitive scintillation detector. The detectors are designed to remain fully operational over a wide range of temperatures. If they are exposed to high radiation transients exceeding the channel range, the channel maintains its operation and returns to normal functioning when the transients have subsided. Since gamma detectors are used, comparison of monitor readout with the results of grab samples is possible. The grab samples are counted in the plant multichannel gamma pulse height spectrometer to check for proper monitor operation. Solenoid-operated check sources are provided to check detector response.

11.5.2.4.3 Steam Generator Blowdown Liquid Monitor: The SG blowdown liquid monitor samples the liquid from either the SG blowdown flash tank or demineralizer. The sample is continuously monitored by a shielded gamma-sensitive detector. Detection of high radiation by this monitor alerts the operator to the possibility of primary-to-secondary leakage.

In the event of activity above the high alarm setpoint or monitor failure, the monitor initiates the automatic closure of FV-5019, the SG blowdown discharge to neutralization basin isolation valve.

11.5.2.4.4 Liquid Waste Processing System Monitors: LWPS monitor no. 1 detects activity present in the liquid waste effluent being discharged from the waste monitor tanks in the LWPS. The monitor is located upstream of the LWPS diversion valve, FV-4077. Upon initiation of a high radiation or monitor failure alarm, the monitor causes the valve to automatically divert the effluent back to the waste monitor tanks.

The sample is drawn from the CCW pump discharge line downstream of the CCW heat exchanger, monitored, and then returned to the CCW surge tank.

11.5.2.4.6 Boron Recycle System Monitor: This monitor is located in the BRS evaporator condensate line downstream of the recycle evaporator condensate filter.

Upon initiation of a high radiation or monitor failure alarm, the monitor initiates changeover of the BRS diversion valve, RCV-4202, causing the BRS condensate to be diverted from the Reactor Makeup Water Storage Tank (RMWST) back to the BRS recycle evaporator feed demineralizers.

11.5.2.4.7 Turbine Generator Building Drain Monitor: This monitor monitors the water in TGB drain sump no. 1. Upon detection of high radiation level or monitor failure, the monitor automatically stops the sump pumps and alarms the condition.

11.5.2.4.8 Failed Fuel Monitor: This monitor takes a sample downstream of the letdown heat exchanger (HX) of the CVCS and acts as a gross-failed fuel detector. The radiation alarms alert the operator to an abnormal increase in gross gamma activity in the CVCS letdown system possibly indicative of fuel cladding failure. Determination of the cause can be made by laboratory analysis. The sample location provides a letdown sample point prior to filtration and demineralization.

11.5.2.4.9 Condensate Polishing System Monitor: This monitor is located on the discharge of the Condensate Polishing System to the neutralization basin. Upon detection of high radiation or monitor failure, the monitor sends a signal to automatically close FV-5804 to terminate discharge to the basin.

11.5.2.5 Adjacent-to-Line Monitors. Adjacent-to-line (ATL) monitors are used to monitor:

1. GWPS

2. MS System

3. Steam Generator Blowdown System (SGBS)

These monitors are mounted adjacent to the process line and do not require a sample stream to monitor for radioactivity.

11.5.2.5.1 Gaseous Waste Processing System Inlet Monitor: The GWPS inlet monitor employs a gamma (NaI crystal) scintillator/photomultiplier tube combination to measure the radioactivity level of the waste gases entering the GWPS. The monitor is used in conjunction with the GWPS discharge monitor to measure overall effectiveness of the GWPS.

11.5.2.5.2 GWPS Discharge Monitor: This monitor is similar to the GWPS inlet monitor and is installed upstream of the GWPS discharge valve. Upon detection of high radioactivity or monitor failure, the GWPS discharge valve, FV-4671, is automatically closed.

11.5.2.5.3 Main Steam Line Monitors: Each MS line is monitored by an ATL monitor consisting of a Geiger Mueller (GM) tube detector and an ion chamber detector with overlapping ranges. The detectors are shielded by 3 in. of lead.

11.5.2.5.1 Gaseous Waste Processing System Inlet Monitor: The GWPS inlet monitor employs a gamma (NaI crystal) scintillator/photomultiplier tube combination to measure the radioactivity level of the waste gases entering the GWPS. The monitor is used in conjunction with the GWPS discharge monitor to measure overall effectiveness of the GWPS.

11.5.2.5.2 GWPS Discharge Monitor: This monitor is similar to the GWPS inlet monitor and is installed upstream of the GWPS discharge valve. Upon detection of high radioactivity or monitor failure, the GWPS discharge valve, FV-4671, is automatically closed.

11.5.2.5.3 Main Steam Line Monitors: Each MS line is monitored by an ATL monitor consisting of a Geiger Mueller (GM) tube detector and an ion chamber detector with overlapping ranges. The detectors are shielded by 3 in. of lead.

The monitors are designed to monitor gross gamma activity in the steam line and provide a basis for determining possible atmospheric releases from the MS power-operated relief valve (PORV), SG safety valves, and/or auxiliary feedwater pump turbine.

The monitors provide a dose rate range equivalent to 10^{-1} to 10^{3} $\mu Ci/cm^3$ xenon-133. Based upon core inventory, the ratio of xenon-133 to other nuclides in the fuel can be detemined. In order to obtain the above concentrations of xenon-133 in the main steam line, a large primary-to-secondary leak must be ~~present coincident with a large amount of fuel failure.~~ The presence of xenon-133 indicates other radioactive isotopes are present.

Using the relative ratios of isotopes present in the MS line, a computer model for determination of dose rates from these isotopes, detector response curves, the thickness of the MS line, and the geometry of the MS line relative to the detector, the dose rate equivalent to MS line concentration is obtained. The quantity of radioactive effluents released is obtained by multiplying the xenon-133 equivalent MS line concentrations by the isotope ratio times the steam release rate.

These detectors are safety-related Class 1E and meet the requirements of RG 1.97 and NUREG-0737.

11.5.2.5.4 Steam Generator Blowdown Monitors: These monitors are identical to the MS line monitors and are adjacent to the SG blowdown lines in the Isolation Valve Cubicle (IVC).

The monitors are used as an aid in determining the source of SG blowdown radioactivity due to SG tube rupture or a large primary-to-secondary leak.

~~These detectors are safety-related Class 1E and meet the requirements of RG 1.97.~~

11.5.2.5.5 Main Steam Line High Energy Gamma (N-16) Monitors: Each main steam line is monitored by an ATL NaI scintillation detector. These detectors were installed to monitor the status of steam generator primary to secondary tube leaks and to provide a diagnostic tool for all individuals concerned with steam generator condition. These detectors are designed to detect high energy gamma activity in the 6 to 7.2 MEV energy range. High energy gamma activity in the main steam lines indicates the presence of N-16. The level of N-16 in the main steam lines is used to

RU2

SFP WATER LEVEL HI/LO

Automatic Actions: None

Immediate Actions: None

Subsequent Actions:

_____ 1) DISPATCH an Operator to the Spent Fuel Pool (SFP) to locally monitor SFP level. {North end of Spent Fuel Pool}

_____ 2) IF any of the following conditions exist, THEN GO TO 0POP04-FC-0001, Loss of Spent Fuel Pool Level or Cooling.

_____ • Unexpected SFP level decrease to less than 66 ft (e.g. other than normal evaporation)

_____ • SFP level is greater than 67 ft.

_____ 3) IF alarm is due to an instrument failure, THEN INITIATE a Condition Report to correct the problem.

Probable Causes:

1) Unisolated SFP makeup or drain path
2) Instrument failure

Origin and Setpoint:

1) 1(2)FC-LSHL-1401 LO: 66 feet elevation (529 inches)
HI: 67 feet elevation (541 inches)

References:

1) P&ID, 5R219F05028
2) Instrument Setpoint Index, 5Z01-0-Z-480001
3) 0POP04-FC-0001, Loss of Spent Fuel Pool Level or Cooling

22M02-F-5

SFP
WATER LVL
HI/LO

SFP TROUBLE

Automatic Actions: None

NOTE

- IF the alarm setpoint of 114°F on 1(2)FC-TIS-1400 is exceeded from the beginning of Core Offload until the completion of Core Reload, THEN the alarm setpoint may be raised to any value less than or equal to the limit specified in the Plant Curve Book Figure 5.15, by using Demand PM IC-1-FC-86003920(IC-2-FC-89003378).
- IF the alarm setpoint is raised, THEN the PM used to raise the setpoint SHALL be scheduled to lower the setpoint to 114°F thirty days after the setpoint was raised.
- IF after thirty days the decay heat load is still NOT low enough to allow returning the setpoint to 114°F, THEN the PM SHALL be rescheduled for performance after another thirty days. This rescheduling process SHALL continue until the alarm setpoint has been returned to 114°F.

Immediate Actions: None

Subsequent Actions: 1) CHECK the following Plant Computer points:

_____ • FCFD1417 (Refueling Water Purification Pump Low Flow)
_____ • FCID1403 (SFP Skimmer Pump Control Power Not Available or Motor Overload)
_____ • FCID1419 (Refueling Water Purification Pump Control Power Not Available or Motor Overload)
_____ • FCQD1406A (SFP Cooling Pump 1A(2A) Trip)
_____ • FCQD1408A (SFP Cooling Pump 1B(2B) Trip)
_____ • FCID1406A (SFP Cooling Pump 1A(2A) Overcurrent Trip)
_____ • FCID1408A (SFP Cooling Pump 1B(2B) Overcurrent Trip)
_____ • FCTD1400 (SFP Water Temperature High)
_____ • FCTD1421 (ICSA Water Temperature Hi)
_____ • FCLC1420 (ICSA Water Level Hi/Lo)
_____ • HFQD9573 (SFP Pump 1A(2A) Room Cooler Control Power Not Available or Motor Overload)
_____ • HFQD9572 (SFP Pump 1B(2B) Room Cooler Control Power Not Available or Motor Overload)
_____ • HFHD9572 FHB FUEL POOL PMP AH21B HS in PTL.
_____ • HFHD9573 FHB FUEL POOL PMP AH21A HS in PTL.

PURPOSE

This procedure provides guidelines to be followed during a low level, high level, or loss of cooling in the Spent Fuel Pool.

SYMPTOMS OR ENTRY CONDITIONS

Spent Fuel Pool water level less than 62 feet (Fuel Pool Level less than 23 feet above the top of irradiated fuel assemblies seated in the storage racks).

Spent Fuel Pool water temperature meets any of the following:

a. Greater than 129°F with all Fuel Assemblies in the Reactor Vessel (Core Fully Loaded).

b. Greater than the limit specified in Plant Curve Book Figure 5.15, Core Offload Spent Fuel Pool Operational Requirement, from the beginning of Core Offload until the completion of Core Reload.

Any of the following Control Room annunciator alarms:

a. "SFP COOLING FLOW LO" Lampbox 22M02 Window E-5

b. "SFP WATER LVL HI/LO" Lampbox 22M02 Window F-5

c. "SFP TROUBLE" Lampbox 22M02 Window F-6

Spent Fuel Pool Gate Seal Air Pressure Low alarm (Local panel by gate)

Rising radiation levels on Spent Fuel Pool Area radiation monitor RT-8090 or RT-8091.

STEP DESCRIPTION FOR 0POP04-FC-0001 Addendum 1 STEP 3.0 and 4.0

STEP:

3.0 INITIATE Actions To Restore Spent Fuel Pool (SFP) Level Per 0POP02-FC-0001, Spent Fuel Pool Cooling And Cleanup System

4.0 CHECK SFP Level Meets The Following:

PURPOSE: To provide direction on methods to fill the Spent Fuel Pool.

Normal makeup to compensate for evaporation should be from Reactor Makeup Water or Demineralized Water. Makeup for leakage should be from Refueling Water Storage Tank or the Boron Recycle System.

If normal sources of Spent Fuel Pool makeup are not available using 0POP02-FC-0001, then a LHSI pump is used to pump borated water from the RWST to the Spent Fuel Pool.

If none of the demineralized water or borated water sources of Spent Fuel Pool makeup are available, OR the leak is very large, then a Fire Water Pump is used to makeup to the Spent Fuel Pool. Use of Fire Water is selected on a large leak due to the potential for rapidly rising radiation levels. Radiation levels on the Fuel Handling Building (FHB) operating deck from direct radiation from the fuel would range from **thousands of rem per hour at the edge of the pool** to **350 R/hr farther away** (no line-of-sight with the racks) near the edge of the pool. The radiation field **near the stairway to the truck bay floor on the southeast corner of the operating deck would be about 300 R/hr.** Radiation levels on the **truck bay floor from radiation scattered from the FHB ceiling would be about 96 R/hr.** These rates assume a full core discharge at 100 hours decay and the Region 2 racks 80% filled with fuel that has decayed for 5 years. (The decayed fuel in the Region 2 racks provides a negligible contribution to the direct doses.) The fuel is assumed to be covered with water. These values do not include the radiation contribution from immersion in a cloud of noble gases (i.e., assumes no clad failure).

The above assumed the fuel is just submerged in water. If the fuel is uncovered, the direct dose from the assemblies would increase somewhat, but not that significantly. The matrix of uranium provides a significant amount of self-shielding. Most of the external dose is due to the peripheral assemblies and the top 12-18 inches of the fuel assemblies. The contribution of the presence or absence of water in the rack is negligible compared to the variability of the assumed assembly sources.

The major impact of uncovering the rack would not be direct radiation, but possible overheating of the clad and subsequent clad failure. If there were widespread clad failure, the noble gases, which are a significant source of the direct radiation, would escape and form a cloud in the FHB. This would significantly increase the quoted dose rates since there is no shielding from the intervening uranium fuel rods.

STEP DESCRIPTION FOR 0POP04-FC-0001 Addendum 1 STEP 6.0

STEP: CHECK SFP Level – GREATER THAN 62 FEET

PURPOSE: At 62 feet, water is at the LCO limit of Technical Specification 3.9.11.1 and Action needs to be taken per that Technical Specification.

BASIS: The top of the Spent Fuel Storage Racks is at 39' 10" per FSAR 9.1.2.2. The top of a fuel assembly (top of top nozzle) in a Spent Fuel Storage Rack is at 38' 10". (Ref. 9N5064, 4480-00054, FSAR Figure 4.2-2B) so 23 feet above the top of an irradiated fuel assembly is at 61' 10".

ACTIONS: TAKE appropriate action per Technical Specification 3.9.11.1.

INSTRUMENTATION: Level Gauge in Spent Fuel Pool

CONTROL/EQUIPMENT: N/A

KNOWLEDGE: N/A

5.6.1.7 The new fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water and less than or equal to 0.98 when filled with aqueous foam moderation (low density water). This requirement shall be met by limiting the fuel assembly nominal enrichments to 5.0 w/o or less.

b. A nominal 21 inches center to center distance between fuel assemblies.

5.6.1.8 The In-containment fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water. This requirement shall be met by satisfying at least one of the following criteria:

1) a maximum initial fuel assembly nominal enrichment to 4.5 w/o or less; or,

2) a minimum number of Integral Fuel Burnable Absorbers (IFBA), as a function of initial nominal assembly enrichment, as shown on Figure 5.6-20, or a K_{inf} of less than or equal to 1.484. The fuel assembly K_{inf} shall be based on a unit assembly configuration (infinite in the lateral and axial extent) in the reactor core geometry, assuming unborated water at 68°F.

The IFBA pin requirements shown in Figure 5.6-20 are based on a nominal IFBA linear B^{10} loading of 1.57 mg-B^{10}/inch. For higher IFBA linear B^{10} loadings, the required number of IFBA pins per assembly may be reduced by the ratio of the increased B^{10} loading to the nominal 1.57 mg-B^{10}/inch loading.

b. A nominal 16 inches center to center distance between fuel assemblies.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 62 feet-6 inches.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1969 fuel assemblies.

The new fuel assemblies are transported to the new fuel storage pit or to the new fuel elevator by the 15/2-ton, dual-service FHB crane. The 2-ton hoist of this crane is designed to handle new fuel assemblies. New fuel handling is discussed in detail in Section 9.1.4. Use of the 2-ton hoist of the 15/2-ton crane or of the fuel-handling machine to handle new fuel ensures that the design uplift of the racks will not be exceeded.

The new fuel storage pit is situated in the approximate center of each FHB. The floor of the new fuel storage pit is at El. 50 ft-3 inches. The new fuel storage pit access hatch is provided with a three-section protective cover at El. 68 ft. The fuel assemblies are loaded into the new fuel storage racks through the top and stored vertically.

9.1.1.3 Safety Evaluation. Units 1 and 2 of the STPEGS are each provided with separate and independent fuel handling facilities.

Flood protection of each FHB is discussed in Section 3.4.1. Flooding of the new fuel storage pit from fluid sources inside either FHB is not considered credible since all fluid systems components are located well below the elevation of the new fuel storage pit access hatch. A floor drain is provided in the new fuel storage pit to minimize collection of water.

The applicable design codes and the ability of the FHB to withstand various external loads and forces are discussed in Section 3.8.4. Details of the seismic design and testing are presented in Section 3.7. Missile protection of the FHBs is discussed in Section 3.5. Failure of nonseismic systems or structures will not decrease the degree of subcriticality provided in the new fuel storage pit.

In accordance with American National Standards Institute (ANSI) N18.2, the design of the normally dry new fuel storage racks is such that the effective multiplication factor will not exceed 0.98 with fuel of the highest anticipated enrichment in place, assuming optimum moderation (under dry or fogged conditions). For the unborated flooded condition, assuming new fuel of the highest anticipated enrichment in place, the effective multiplication factor does not exceed 0.95. Credit may be taken for the inherent neutron-absorbing effect of the materials of construction.

The new fuel assemblies are stored dry, the 21-in. spacing ensuring a safe geometric array. Under these conditions, a criticality accident during refueling and storage is not considered credible. Consideration of criticality safety analysis is discussed in Section 4.3.

Design of the facility in accordance with RG 1.13 ensures adequate safety under both normal and postulated accident conditions. The new fuel storage racks also meet the requirements of General Design Criterion (GDC) 62.

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases. The spent fuel pool (SFP) is a stainless steel-lined reinforced concrete pool and is an integral part of each FHB. All spent fuel racks are classified as seismic Category I, as defined by RG 1.29, and as ANS SC 3.

The spent fuel storage facility provides storage capacity for 1,969 high density absorber spent fuel racks in a honeycomb array in each unit. Two storage regions are provided in the SFP. Two of the

Region 2 rack modules on the south end of the pool (modules #12 and #16) have not been installed. A Fuel Ultrasonic Cleaning system may be used in the open space designated for modules #12 and #16. The Fuel Ultrasonic Cleaning system is freestanding and is seismically qualified. It has no adverse effect on the fuel assemblies that are selected for cleaning; nor does it have an adverse effect on the design function of the spent fuel pool or its associated support systems. Figure 9.1.2-2 shows the pool layout for both Units 1 and 2. The six Region 1 rack modules are located in the northwest corner of the spent fuel pool.

The Region 1 racks have 10.95-in. nominal center-to-center spacing between the cells. This region is conservatively designed to accommodate unirradiated fuel at enrichments to 4.95 weight percent. Region 1 storage cells are each bounded on four sides by a water box except on the periphery of the pool. The Region 1 spent fuel racks include a lead-in-guide to assist in depositing fuel assemblies into the fuel cell. Figure 9.1.2-3 shows a typical Region 1 spent fuel rack.

The reactivity characteristics of fuel assemblies which are to be placed in the spent fuel storage racks are determined and the assemblies are categorized by reactivity. Alternately, if necessary, all assemblies may be treated as if each assembly is of the highest reactivity class until the actual assembly reactivity classification is determined. Section 5.6 of the Technical Specifications provides the definitions of the reactivity classifications and the allowed storage patterns. Fuel assemblies are loaded into the racks in a geometrically safe configuration to ensure rack subcriticality.

Fuel assembly reactivity requirements for close packed storage and checkerboard storage are specified in the Technical Specifications. The boron concentration of the water in the spent fuel pool is maintained at or above the minimum value needed to ensure that the rack K_{eff} is less than or equal to 0.95 in the event of misplaced assemblies in the close packed storage areas or in checkerboard storage areas. Consideration of criticality safety is discussed in Section 4.3.

The Region 2 racks have a 9.15-in. nominal center-to-center spacing with fixed absorber material surrounding each cell. A sheet of neutron absorber material is captured between the side walls of all adjacent boxes. To provide space for the absorber sheet between boxes, a double row of matching flat round raised areas are coined into the side walls of all boxes. The raised dimension of these locally formed areas on each box wall is half the thickness of the absorber sheet. The boxes are fusion welded together at all these local areas. The absorber sheets are scalloped along their edges to clear these areas. Figure 9.1.2-4 shows a typical Region 2 spent fuel rack.

The axial location of the absorber with respect to the active fuel region is provided and maintained by the structure of each box. At the outside periphery of each rack, a sheet of absorber material is captured under thin stainless sheets which are intermittently welded all around to the box.

All rack surfaces that come into contact with fuel assemblies are made of annealed austenitic stainless steel. These materials are resistant to corrosion during normal and emergency water quality conditions. The racks are designed to withstand normal operating loads as well as to remain functional with the occurrence of an SSE. The racks are designed with adequate energy absorption capabilities to withstand the impact of a dropped spent fuel assembly from the maximum lift height of the spent fuel pit bridge hoist. The racks are designed to withstand a maximum uplift force equal to the uplift force of the bridge hoist. The 14-in. and 16-in. racks also meet the requirements of ASME Code, Section III, Appendix XVII. The high-density spent fuel racks meet the criteria of Appendix D to Standard Review Plan (SRP) 3.8.4.

Shielding for the SFP is adequate to protect plant personnel from exposure to radiation in excess of published guideline values as stated in Section 12.1. A minimum depth of approximately 13 ft of water over the top of an array of 193 (full core) assemblies with 42 hours of decay is required to limit radiation from the assemblies to 2.5 mR/hr. or less.

The FHB Ventilation Exhaust System is designed to limit the offsite dose in the event of a significant release of radioactivity from the fuel, as discussed in Sections 12.3.3, 15.7.4, and 9.4.2. However, no credit for the FHB Ventilation Exhaust System is taken in the LOCA and Fuel Handling accident in Chapter 15.

The FHB is designed to prevent missiles from contacting the fuel. A more detailed discussion on missile protection is given in Section 3.5.

In addition, space is provided for storage of fuel during refueling inside the RCB for 64 fuel assemblies in four 4 x 4 modules having 16-in. center-to-center spacing (Figure 9.1.2-1A). These modules are firmly bolted in the floor.

9.1.2.2 Facilities Description. The FHB abuts the south side of the RCB and is adjacent to the west side of the MEAB of each unit. The locations of the two FHBs are shown in the station plot plan on Figure 1.2-3. For general arrangement drawings of the spent fuel storage facilities, refer to Figures 1.2-39 through 1.2-48 as listed in Table 1.2-1.

The spent fuel storage facilities are designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor vessel. The spent fuel is transferred to the FHB and handled and stored in the spent fuel pool underwater. The fuel is stored to permit some decay, then transferred offsite. For a detailed discussion of spent fuel handling, see Section 9.1.4.

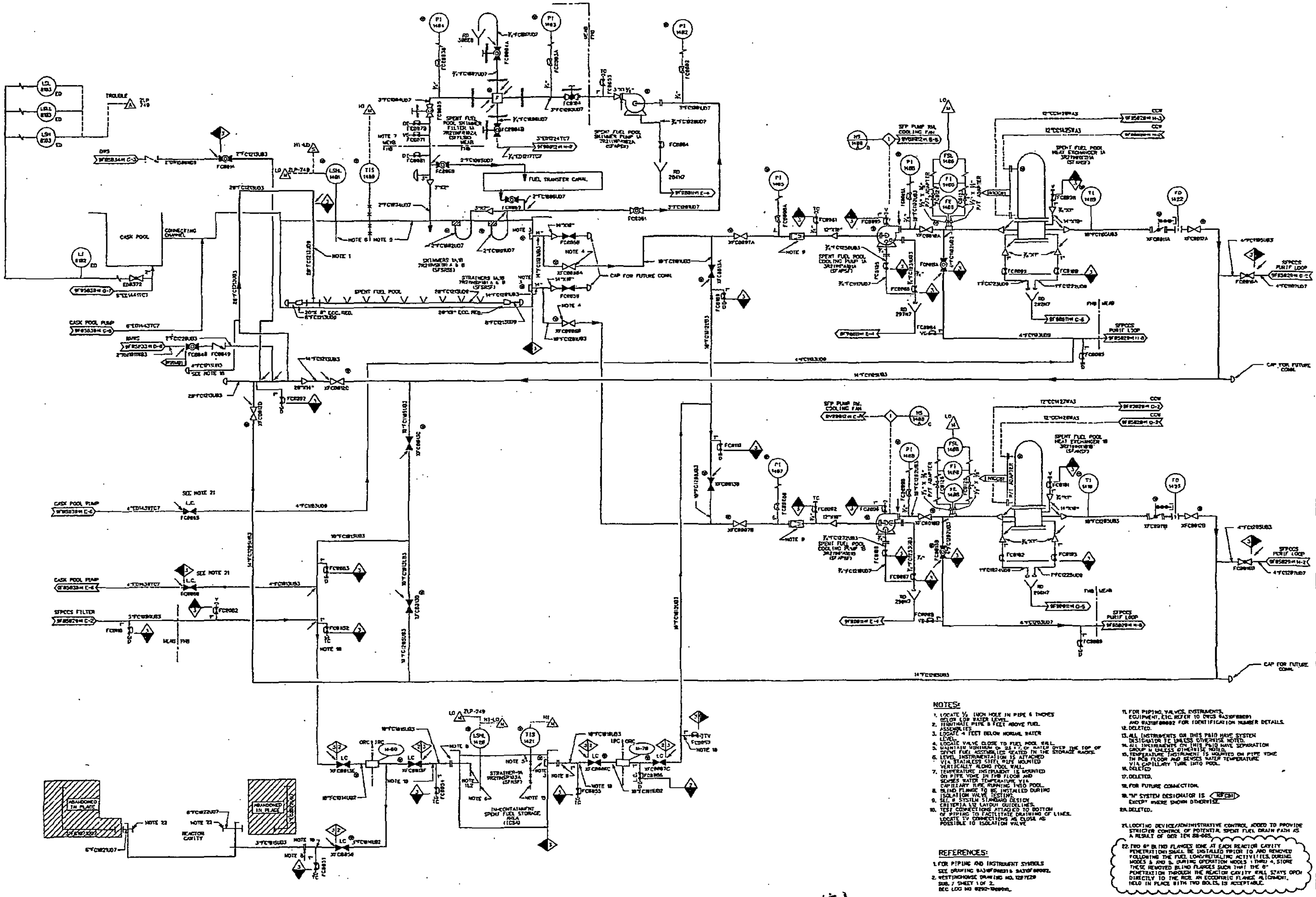
The SFP is located in the northwest quadrant of each FHB. The floor of the pool is at El. 21 ft-11 in., with normal water level at El. 66 ft-6 inches. The top of a fuel assembly in a storage rack does not extend above the top of the storage rack which is El. 39 ft-10 in. maximum. The fuel assemblies are loaded into the spent fuel racks through the top and are stored vertically.

9.1.2.3 Safety Evaluation. Units 1 and 2 of the STPEGS are each provided with separate and independent fuel handling facilities. Flood protection of each FHB is discussed in Section 3.4.1. A detailed discussion of missile protection is provided in Section 3.5.

The applicable design codes and the various external loads and forces considered in the design of the FHB are discussed in Section 3.8.4. Details of the seismic design and testing are presented in Section 3.7.

Design of this storage facility in accordance with GDC 62 and RG 1.13 ensures a safe condition under normal and postulated accident conditions. The K_{eff} of the spent fuel storage racks is maintained less than or equal to 1.00, even if unborated water is used to fill the spent fuel storage pool, by both the designs of the fuel assemblies and the storage rack and the use of administrative procedures to control the placement of burned and fresh fuel and control rods.

Under accident conditions, the K_{eff} is maintained well below 0.95 assuming 2200 ppm borated water. The boron concentration of the water in the spent fuel pool is maintained at or above the minimum



NOTES:

1. LOCATE ½ INCH HOLE IN PIPE 6 INCHES BELOW LOW WATER LEVEL.
2. TERMINATE PIPE 8 FEET ABOVE FUEL ASSEMBLIES
3. LOCATE 4 FEET BELOW NORMAL WATER LEVEL.
4. LOCATE VALVE CLOSE TO FUEL POOL WALL.
5. MAINTAIN MINIMUM OF 25 FT. OF WATER OVER THE TOP OF SPENT FUEL ASSEMBLIES SEATED IN THE STORAGE RACKS.
6. LEVEL INSTRUMENTATION IS ATTACHED VIA STAINLESS STEEL PIPE MOUNTED VERTICALLY ALONG POOL WALL.
7. TEMPERATURE INSTRUMENT IS MOUNTED ON PIPE YOKE IN FHB FLOOR AND SENSES WATER TEMPERATURE VIA CAPILLARY TUBE RUNNING INTO POOL.
8. BLIND FLANGE TO BE INSTALLED DURING ISOLATION VALVE TESTING.
9. SEE # SYSTEM STANDARD DESIGN CRITERIA 1.12 LAYOUT GUIDELINES.
10. TEST CONNECTIONS ATTACHED TO BOTTOM OF PIPING TO FACILITATE DRAINING OF LINES. LOCATE TV CONNECTIONS AS CLOSE AS POSSIBLE TO ISOLATION VALVE
11. FOR PIPING, VALVES, INSTRUMENTS, EQUIPMENT, ETC. REFER TO DWGS 9A359F00001 AND 9A359F00002 FOR IDENTIFICATION NUMBER DETAILS.
12. DELETED.
13. ALL INSTRUMENTS ON THIS P&ID HAVE SYSTEM DESIGNATOR FC UNLESS OTHERWISE NOTED.
14. ALL INSTRUMENTS ON THIS P&ID HAVE SEPARATION GROUP N UNLESS OTHERWISE NOTED.
15. TEMPERATURE INSTRUMENT IS MOUNTED ON PIPE YOKE IN RCB FLOOR AND SENSES WATER TEMPERATURE VIA CAPILLARY TUBE INTO POOL.
16. DELETED
17. DELETED.
18. FOR FUTURE CONNECTION.
19. "N" SYSTEM DESIGNATOR IS NFC001 EXCEPT WHERE SHOWN OTHERWISE.
20. DELETED.
21. LOCKING DEVICE/ADMINISTRATIVE CONTROL ADDED TO PROVIDE STRICTER CONTROL OF POTENTIAL SPENT FUEL DRAIN PATH AS A RESULT OF DER 1EN 88-065.
22. TWO 6" BLIND FLANGES (ONE AT EACH REACTOR CAVITY PENETRATION) SHALL BE INSTALLED PRIOR TO AND REMOVED FOLLOWING THE FUEL LOAD/REFUELING ACTIVITIES DURING MODES 5 AND 6. DURING OPERATION MODES 1 THRU 4, STORE THESE REMOVED BLIND FLANGES SUCH THAT THE 6" PENETRATION THROUGH THE REACTOR CAVITY WALL STAYS OPEN DIRECTLY TO THE RCB. AN ECCENTRIC FLANGE ALIGNMENT, HELD IN PLACE WITH TWO BOLTS, IS ACCEPTABLE.

REFERENCES:

1. FOR PIPING AND INSTRUMENT SYMBOLS SEE DRAWING 9A369F00001 & 9A369F00002.
2. WESTINGHOUSE DRAWING NO. 1287E20 SUB. 7 SHEET 1 OF 2. BEC LOG NO 0292-[illegible]

DRAWING 5R219FØ5028 #1

28	10-24-11	INCORP. DCN 1101478.	VJM	[illegible]	N/A	N/A	N/A	N/A
27	10-13-04	INCORP. DCN 0301847	KS	ON FILE				
26	8-21-99	INCORP. DCN 9981795	JSH	ON FILE				
25	3-16-99	INCORP. DCN 9803996.	SIGNATURES ON FILE					
24	4-6-83	ISSUED FOR CONSTRUCTION	SIGNATURES ON FILE					

SOUTH TEXAS PROJECT
NUCLEAR OPERATING COMPANY

PIPING & INSTRUMENTATION DIAGRAM
SPENT FUEL POOL
COOLING & CLEANUP SYSTEM

PRIORITY	CADD FILE NO.	FSUG. NO.	STI. NO	SCALE	DWG NO	SHT.	REV.

TABLE 12.3.4-1

AREA RADIATION MONITORS

Reactor Containment Building

Tag Number and Location [1]	Range (mR/hr) [3]	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8052 Incore Instrumentation Room (-1ft-6 in.)	10^{-1}-10^{4}	1,000
N1RA-RE-8053 Support across from elevator (-11 ft-3 in.)	10^{-1}-10^{4}	100
N1RA-RE-8054 West Stair Landing (19 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8055 North SG wall across from the head laydown area (68 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8056 Support across from elevator (52 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8099 South SG wall across from the in-containment fuel pool (68 ft-0 in.)	10^{-1}-10^{4}	100

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.
2. The alarm setpoints listed are typical and may be varied as necessary.
3. Range is based on microprocessor conversion factor and a detector signal which has a high degree of confidence. Conversion factor will vary dependent on the detector calibration. Exact ranges are found in plant instrument scaling manuals.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Fuel Handling Building

Tag Number and Location [1]	Range (mR/hr)	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8081 ~11 ft S of cols. 30.2 and S_5 (68 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8084 ~24 ft S of cols. 28 and T_5 (-21 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8085 ~24 ft S of col. 28 and ~6 ft E of col. S_5 (-21 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8086 ~24 ft S of col. 28 and ~11 ft E of col. R_1 (-21 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8087 col. 30.2 and 12 ft W of col. R_1 (4 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8088 3 ft S of col. 30.9 and col. R_1 (30 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8089 col. 28 and col. N (68 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8090 18 ft N of col. 30.2 and col. T_5 (68 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8091 col. 34 and col. N (68 ft-0 in.)	10^{-1}-10^{4}	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Fuel Handling Building (Continued)

Tag Number and Location [1]	Range (mR/hr)	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8097 33 ft S of cols. 28 and 10 ft W of col. N (68 ft-0 in.)	10^{-2}-10^{7}	1,000

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building

Tag Number and Location [1]	Range (mR/hr) [3]	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8057 col. 22 and ~10 ft E of col. J (10 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8058 col. 26 and col. J (10 ft-0 in.)	10^{-1}-10^{4}	2.5
NIRA-RE-8059 col. 27 and col G (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8060 ~10 ft S of col. 30 and col. E (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8061 ~10 ft S of col. 24 and ~11 ft W of col. E (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8062 ~6 ft S of col. 31 and col. C (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8063 ~9 ft S of col. 28 and col. B (10 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8064 ~12 ft S of col. 24 and col. F (29 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8065 ~5 ft N of col. 32 and col. C (29 ft-0 in.)	10^{-1}-10^{4}	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

3. Range is based on microprocessor conversion factor and a detector signal which has a high degree of confidence. Conversion factor will vary dependent on the detector calibration. Exact ranges are found in plant instrument scaling manuals

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building (Continued)

Tag Number and Location [1]	Range (mR/hr) [3]	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8066 ~4 ft N of col. 22 and 14 ft E of col. C (35 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8067 col. 22 and 10 ft E of col. J (35 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8068 ~10 ft N of col. 25 and col. H (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N2RA-RE-8068 ~10 ft S of col. 24 and col. G (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8069 ~12 ft S of col. 24 and ~14 ft E of col. C (41 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8070 col. 29 and col. C (41 ft-0 in.)	10^{-2}-10^{3}	2.5
N1RA-RE-8071 ~18 ft S of col. 28 and 3 ft W of col. B (41 ft-0 in.)	10^{-2}-10^{3}	2.5
N1RA-RE-8072 ~11 ft N of col. 29 and 5 ft W of col. D (41 ft-0 in.)	10^{-1}-10^{4}	100
N1RA-RE-8073 col. 29 and col. E (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8074 ~5 ft S of col. 31 and ~7 ft W of col. C (41 ft-0 in.)	10^{-1}-10^{4}	2.5

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Mechanical Electrical Auxiliaries Building (Continued)

Tag Number and Location (1)	Range (mR/hr)	High Alarm Setpoint (mr/hr) (2)
N1RA-RE-8075 col. 28 and ~3 ft W of col. G (41 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8076 col. 22 and ~10 ft E of col. J (60 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8077 col. 27 and col. J (60 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8078 col. 27 and col. F (60 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8079 col. 25 and ~2 ft W of col. F (60 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8080 col. 26 and col. H (41 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8082 col. 28 and ~8 ft E of col. H (69 ft-0 in.)	10^{-1}-10^{4}	2.5
N1RA-RE-8083 ~10 ft S of col. 29 and 15 ft W of col. E (41 ft-0 in.)	10^{-1}-10^{4}	15.0
N1RA-RE-8098 ~6 ft N of col. 25 and col. H (60 ft-0 in.)	10^{2}-10^{7}	1000

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Miscellaneous Buildings

Tag Number and Location [1]	Range (mR/hr)	High Alarm Setpoint (mr/hr) [2]
N1RA-RE-8092 col. 9 and col. P TGB (29 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8093 col. 7 and col. M TGB (29 ft-0 in.)	10^{-2}-10^{3}	0.5
N1RA-RE-8094 ~3 ft N of col. 23 and ~14 ft W of col. B TSC-MEAB (72 ft-0 in.)	10^{-2}-10^{7}	1000

CN-2963

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.

2. The alarm setpoints listed are typical and may be varied as necessary.

TABLE 12.3.4-1 (Continued)

AREA RADIATION MONITORS

Post-Accident Monitors

Tag Number and Location [1]	Range (R/hr)	High Alarm Setpoint (R/hr) [2]
A1RA-RE-8050 RCB (68 ft-0 in.)	10^0-10^8	2000
C1RA-RE-8051 RCB (68 ft-0 in.)	10^0-10^8	2000

1. Tag Number is shown for Unit 1. For Unit 2, change second position from 1 to 2.
2. The alarm setpoints listed are typical and may be varied as necessary.

RA1

ENERCON Excellence—Every project. Every day.	CALCULATION COVER SHEET	CALC. NO.STPNOC013-CALC-002 REV. 1 PAGE NO. 1 of 49

Title:	Radiological Release Thresholds for Emergency Action Levels	**Client:** South Texas Project	**Project:** STPNOC013

Item	Cover Sheet Items	Yes	No
1	Does this calculation contain any open assumptions that require confirmation? (If **YES**, Identify the assumptions) ______		✓
2	Does this calçulation serve as an "Alternate Calculation"? (If **YES**, Identify the design verified Calculation.) **Design Verified Calculation No.** ______		✓
3	Does this calculation Supersede an existing Calculation? (If **YES**, identify the superseded Calculation.) **Superseded Calculation No.** ______		✓

Scope of Revision: Incorporate decay time of one hour from shutdown as well as migration into Attachment 1. Change statement of no decay in the STAMPEDE runs.

Revision Impact on Results: Values calculated in Attachment 1 decreased and have become the limiting values.

Study Calculation ☐ **Final Calculation** ☑

Safety-Related ☑ **Non-Safety Related** ☐

(Print Name and Sign)

Originator:	Caleb Trainor	**Date:** 3/21/2014
Design Verifier:	Chad Cramer	**Date:** 3/21/14
Approver:	Marvin Morris	**Date:** 3/21/14

ENERCON *Excellence—Every project. Every day.*	CALCULATION REVISION STATUS SHEET	CALC. NO. STPNOC013-CALC-002
		REV. 1
		PAGE NO. 2 of 49

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
0	03/03/2014	Initial Issue
1	3/21/2014	Resolve inconsistency in decay times for the two calculations

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION
1-11	1		

ATTACHMENT REVISION STATUS

ATTACHMENT NO.	PAGE NO.	REVISION NO.	ATTACHMENT NO.	PAGE NO.	REVISION NO.
1	12-24	1			
2	25-31	1			
3	32-49	1			

ENERCON Excellence—Every project. Every day.	CALCULATION DESIGN VERIFICATION CHECKLIST	CALC. NO. STPNOC013-CALC-002 REV. 1 PAGE NO. 3 of 49

Item	CHECKLIST ITEMS	Yes	No	N/A
1	**Design Inputs** - Were the design inputs correctly selected, referenced (latest revision), consistent with the design basis and incorporated in the calculation?	✓		
2	**Assumptions** – Were the assumptions reasonable and adequately described, justified and/or verified, and documented?	✓		
3	**Quality Assurance** – Were the appropriate QA classification and requirements assigned to the calculation?	✓		
4	**Codes, Standard and Regulatory Requirements** – Were the applicable codes, standards and regulatory requirements, including issue and addenda, properly identified and their requirements satisfied?	✓		
5	**Construction and Operating Experience** – Have applicable construction and operating experience been considered?			✓
6	**Interfaces** – Have the design interface requirements been satisfied, including interactions with other calculations?	✓		
7	**Methods** – Was the calculation methodology appropriate and properly applied to satisfy the calculation objective?	✓		
8	**Design Outputs** – Was the conclusion of the calculation clearly stated, did it correspond directly with the objectives and are the results reasonable compared to the inputs?	✓		
9	**Radiation Exposure** – Has the calculation properly considered radiation exposure to the public and plant personnel?	✓		
10	**Acceptance Criteria** – Are the acceptance criteria incorporated in the calculation sufficient to allow verification that the design requirements have been satisfactorily accomplished?	✓		
11	**Computer Software** – Is a computer program or software used, and if so, are the requirements of CSP 3.02 met?			✓

COMMENTS:
None

	(Print Name and Sign)	
Design Verifier:	Chad Cramer [signature]	**Date:** 3/21/14
Others:		**Date:**

<table>
<tr><td rowspan="3">ENERCON
Excellence—Every project. Every day.</td><td rowspan="3">CALCULATION
DESIGN VERIFICATION
PLAN AND SUMMARY SHEET</td><td>CALC. NO. STPNOC013-CALC-002</td></tr>
<tr><td>REV. 1</td></tr>
<tr><td>PAGE NO. 4 of 49</td></tr>
</table>

Calculation Design Verification Plan:

Calculation shall be verified by comparing the documented input with the references and checking the validity of the references for the intended use. As necessary, assumptions shall be evaluated and verified to determine if they are based on sound engineering principles and practices. Verify the applicable methodology, inputs, results, and conclusions.

(Print Name and Sign for Approval – mark "N/A" if not required)

Approver: Marvin Morris Marvin Morris	**Date** 3/21/14

Calculation Design Verification Summary:

Design inputs, assumptions, methodology, results and conclusions were evaluated/verified and found to be acceptable. All comments have been incorporated.

Based On The Above Summary, The Calculation Is Determined To Be Acceptable.

(Print Name and Sign)

Design Verifier: Chad Cramer	**Date:** 3/21/14
Others:	**Date:**