Nuclear Operating Company
South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

July 1, 2015
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10 CFR 50.90
File No. G25

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

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Response to Request for Additional Information and Supplement to
South Texas Project (STP), Units 1 and 2 License Amendment Request for
Emergency Action Level Scheme Change (TACs MD4195 and MF4196)

References:

1. Letter; G. T. Powell to USNRC Document Control Desk; "License Amendment Request for Revision to Unit 1 and Unit 2 Emergency Action Levels;" NOC-AE-14003087; dated May 15, 2014 (ML14164A341)

2. Letter; A. Capristo to USNRC Document Control Desk; "Response to Request for Additional Information - South Texas Project (STP), Units 1 and 2 License Amendment Request for Emergency Action Level Scheme Change (TACs MD4195 and MF4196);" NOC-AE-15003214; dated February 11, 2015 (ML15055A039)

3. Letter; A. Capristo to USNRC Document Control Desk; "Response to Request for Additional Information - South Texas Project (STP), Units 1 and 2 License Amendment Request for Emergency Action Level Scheme Change (TACs MD4195 and MF4196);" NOC-AE-15003226; dated February 26, 2015 (ML15068A045)

4. Letter; L. Regner to D. Koehl; "South Texas Project, Units 1 And 2 - Request for Additional Information, License Amendment Request to Revise the Emergency Action Level Scheme to the NRC-endorsed Scheme Contained in NEI-99, Revision 6 (TAC Nos. MF4195 and MF4196);" dated April 15, 2015 (ML15100A007)

By Reference 1, as supplemented by References 2 and 3, STP Nuclear Operating Company (STPNOC) requested approval of a License Amendment Request for revision to Unit 1 and 2 Emergency Action Levels. By Reference 4, the NRC staff requested additional information (RAI) to complete its review. STPNOC's response to Reference 4 is provided in Attachment 1 to this letter.

While addressing the staff's RAI, STPNOC identified errors in the calculation for the fission product barrier Emergency Action Level (EAL) thresholds and generated a condition report in the station Corrective Action Program. A new calculation (15-RA-011) was created for

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calculating the fission product barrier EAL threshold. The new calculation replaces calculation STPNOC013-CALC-004 which is referenced in the original submittal. Also, a support calculation (03-ZE-003) used for determining contingency monitor thresholds for post-accident failed fuel monitoring was revised. Copies of these calculations are provided in Attachment 5 to this letter.

An extent of condition was performed on all other EAL support calculations and STPNOC discovered non-conservative discrepancies in the supporting calculation (STPNOC013-CALC-006) for EALs CS1 and CG1. A revision was made to the calculation; EAL setpoints were not affected, but the EAL bases discussions for CS1 and CG1 have been appropriately revised. Specifically, clarifications were made to the CS1 and CG1 bases to correct the Reactor Coolant System (RCS) level corresponding to the top of the active fuel and the calculated monitor response when postulated RCS level is at the top of the active fuel. A clarification was also made to these bases to indicate that the threshold value is calculated with the reactor head removed as opposed to the reactor head installed. A copy of the revised STPNOC13-CALC-006 calculation is provided in Attachment 5 to this letter.

A clean copy and a redline markup of the STPEGS Emergency Action Level Technical Bases Document is included in Attachments 2 and 3, respectively. Attachment 4 provides revisions to the STPEGS Emergency Action Level Deviation, Difference and Justification Matrix provided in Reference 1.

The No Significant Hazards Consideration determination provided in Reference 1 has been reviewed and determined to remain applicable as previously submitted.

There are no commitments in this letter.

If there are any questions, please contact Drew Richards at (361) 972-7666 or me at (361) 972-7566.

I declare under penalty of perjury that the foregoing is true and correct.

G. T. Powell

Executed on July 1, 2015
Date

G. T. Powell
Site Vice President

amr

Attachment:

1. Response to Request for Additional Information - South Texas Project (STP), Units 1 and 2 License Amendment Request for Emergency Action Level Scheme Change
2. STPEGS Emergency Action Level Technical Bases Document – clean copy
3. STPEGS Emergency Action Level Technical Bases Document – redline markup
4. STPEGS Emergency Action Level Deviation, Difference and Justification Matrix – revisions only
5. Support Documents

cc:

(paper copy)

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Attachment 1

Response to Request for Additional Information - South Texas Project (STP), Units 1 and 2 License Amendment Request for Emergency Action Level Scheme Change

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST FOR EMERGENCY ACTION LEVEL SCHEME CHANGE

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

By letter dated May 15, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML 14164A341), as supplemented by letters dated July 10, 2014, and February 11 and 26, 2015 (ADAMS Accession Nos. ML 14282A185, ML 15055A039, and ML 15068A045, respectively), STP Nuclear Operating Company (STPNOC, the licensee) requested approval for an emergency action level (EAL) scheme change for the South Texas Project Units, 1 and 2.

The licensee's requested changes support a conversion from the current EAL scheme to a scheme based on Nuclear Energy Institute (NEI) 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," dated November 2012 (ADAMS Accession No. ML 12326A805). The following request for additional information (RAI) is necessary for the U.S. Nuclear Regulatory Commission (NRC) staff to complete its review.

General discussion

By letter dated February 26, 2015, STPNOC responded to RAI-10, Attachment 1, page 4 of 5 as follows:

> STPNOC has removed containment radiation monitors (RT-8050 and RT-8051) from the RCS [Reactor Coolant System] Barrier Category 3, RCS Activity/Containment Radiation table due to a reduction in the setpoint value based on calculation STPNOC013-CALC-004, Revision 2. Calculation STPNOC013-CALC-004 was revised in February 2015 and would have lowered the RT-8050 and RT-8051 setpoint from 450 mR/hr to approximately 140 mR/hr above background. STPNOC believes that the proximity of the new setpoint to the background level and the effect of TIC [temperature-induced current] precludes the use of these radiation monitors as reliable indications of an RCS breach. STPNOC does not have other Reg. Guide 1.97 radiation monitors in the containment that can fulfill the function of RT-8050 and RT-8051.

The NEI 99-01, Revision 6, guidance associated with RCS barrier loss (based on section "RCS Activity/Containment Radiation" in Table 9-F-3: PWR [Pressurized-Water Reactor] EAL Fission Product Barrier Table) on page 98 states the following threshold for loss or potential loss of barriers is needed:

> Containment radiation monitor reading greater than (site-specific value)

The loss of RCS barrier based on RCS Activity/Containment Radiation is included to provide a diverse indication that the RCS barrier has been lost. The NRC staff expects licensees to

provide site-specific indications that will promote timely and accurate assessment of barrier status.

RAI-10.a

STPNOC proposes removing the loss of RCS barrier due to Category 3, RCS Activity/ Containment Radiation from the EAL scheme.

Typically, the value associated with this threshold is based upon the maximum allowed amount of fuel damage. Radiation levels above this threshold would therefore be indicative of a loss of the RCS barrier and would require an EAL as determined from the fission barrier logic. It is unusual for this value to be as low as described in STPNOC's response, as compared to licensees of a similar design, which, in some instances, can be a threshold as high as 25 R/hr.

a. Please provide justification that the calculated value is correct. If a discrepancy is identified, provide a correct value.

b. Please provide a containment radiation value that would provide an indication of RCS barrier integrity.

STP RESPONSE RAI-10.a.

a. STP has performed an additional independent analysis of calculation STPNOC013-CALC-004 and identified an underlying assumption that skewed STP's analytical result. STP has replaced STPNOC013-CALC-004 with new calculation 15-RA-011 and has revised the values of the Fission Product Barrier table accordingly. Changes to the assumptions in 15-RA-011 affected calculation 03-ZE-003 and requires a revision to HATCH MONITOR values used in the same Fission Product Barrier EALs. Additionally, the new radiation monitor values overwhelm the influence of the Temperature Induced Current (TIC) and that phenomena information will be removed.

STP's revision to the Fission Product Barrier Table EALs are as follows:

3. RCS Activity / Containment Radiation - Fuel Clad Barrier Loss
A1. RCB Rad Monitor RT-8050 or RT-8051 greater than 2100 R/hr
OR
2. HATCH MONITOR greater than 4200 mR/hr

3. RCS Activity / Containment Radiation - RCS Barrier Loss
A1. RCB Rad Monitor RT-8050 or RT-8051 greater than 10 R/hr
OR
2. HATCH MONITOR greater than 20 mR/hr

3. RCS Activity / Containment Radiation - Containment Barrier Potential Loss
A1. RCB Rad Monitor RT-8050 or RT-8051 greater than 45,000 R/hr
OR
2. HATCH MONITOR greater than 90,000 mR/hr

b. A loss of RCS barrier is indicated by RCB Rad Monitor RT-8050 or RT-8051 reading greater than 10 R/hr or the HATCH MONITOR reading greater than 20 mr/hr.

REFERENCES

1. Calculation 15-RA-011 Rev. 0, Fission Product Barrier Failure for Emergency Action Level Thresholds.
2. Calculation 03-ZE-003 Rev. 1, RT8050/RT8051 Contingency Conversion Constant for Post-Accident Failed Fuel Monitoring.

Attachment 2

STPEGS Emergency Action Level Technical Bases Document – clean copy

STPEGS Emergency Action Level Technical Bases Document Rev. 0

NEI 99-01 Rev. 6 Implementation

June 2015

NOTE: Changes to this document require a review under 10CFR50.54 (q) as directed by 0PGP05-ZV-0010, Emergency Plan Change.

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1 DEVELOPMENT OF EMERGENCY ACTION LEVELS

1.1 REGULATORY BACKGROUND

Title 10, Code of Federal Regulations (CFR), Energy, contains the U.S. Nuclear Regulatory Commission (NRC) regulations that apply to nuclear power facilities. Several of these regulations govern various aspects of an emergency classification scheme. A review of the relevant sections listed below will aid the reader in understanding the key terminology provided in Section 3.0 of this document.

- 10 CFR § 50.47(a)(1)(i)
- 10 CFR § 50.47(b)(4)
- 10 CFR § 50.54(q)
- 10 CFR § 50.72(a)
- 10 CFR § 50, Appendix E, IV.B, Assessment Actions
- 10 CFR § 50, Appendix E, IV.C, Activation of Emergency Organization

The above regulations are supplemented by various regulatory guidance documents. Three documents of particular relevance to NEI 99-01 are:

- NUREG-0654/FEMA-REP-1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, October 1980. [Refer to Appendix 1, Emergency Action Level Guidelines for Nuclear Power Plants]

- NUREG-1022, Event Reporting Guidelines 10 CFR § 50.72 and § 50.73

Regulatory Guide 1.101, Emergency Response Planning and Preparedness for Nuclear Power Reactor

1.2 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

South Texas Project Electrical Generating Station (STP or STPEGS) is locating an ISFSI approximately 450 feet west of the Unit 2 Reactor Building. The STP ISFSI will be within the site Protected Area and is scheduled to be operational in 2016.

Selected guidance in NEI 99-01 is applicable to the STPEGS emergency plan to fulfill the requirements of 10 CFR 72.32 for a stand-alone ISFSI. The emergency classification levels applicable to an ISFSI are consistent with the requirements of 10 CFR § 50 and the guidance in NUREG 0654/FEMA-REP-1. The initiating conditions germane to a 10 CFR § 72.32 emergency plan (as described in NUREG-1567) are subsumed within the classification scheme for a 10 CFR § 50.47 emergency plan.

The STPEGS ICs and EALs for an ISFSI are presented in Section 8, ISFSI ICs/EALs. IC E-HU1 covers the spectrum of credible natural and man-made events included within the scope of the STPEGS ISFSI design. In addition, appropriate aspects of IC HU1 and IC HA1 address a HOSTILE ACTION directed against the STPEGS ISFSI.

The analysis of potential onsite and offsite consequences of accidental releases associated with the operation of an ISFSI is contained in NUREG-1140, *A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees.* NUREG-1140 concluded that the postulated worst-case accident involving an ISFSI has insignificant consequences to public health and safety. This evaluation shows that the maximum offsite dose to a member of the public due to an accidental release of radioactive materials would not exceed 1 rem Effective Dose Equivalent.

Regarding the above information, the expectations for an offsite response to an ALERT classified under a 10 CFR § 72.32 emergency plan are generally consistent with those for an UNUSUAL EVENT in a 10 CFR § 50.47 emergency plan (e.g., to provide assistance if requested). Also, the STPEGS Emergency Response Organization (ERO) required for a 10 CFR § 72.32 emergency plan is different than that prescribed for a 10 CFR § 50.47 emergency plan (e.g., no emergency technical support function).

1.3 NRC ORDER EA-12-051

The Fukushima Daiichi accident of March 11, 2012, was the result of a tsunami that exceeded the plant's design basis and flooded the site's emergency electrical power supplies and distribution systems. This caused an extended loss of power that severely compromised the key safety functions of core cooling and containment integrity, and ultimately led to core damage in three reactors. While the loss of power also impaired the spent fuel pool cooling function, sufficient water inventory was maintained in the pools to preclude fuel damage from the loss of cooling.

Following a review of the Fukushima Daiichi accident, the NRC concluded that several measures were necessary to ensure adequate protection of public health and safety under the provisions of the backfit rule, 10 CFR 50.109(a)(4)(ii). Among them was to provide each spent fuel pool with reliable level instrumentation to significantly enhance the ability of key decision-makers to allocate resources effectively following a beyond design basis event. To this end, the NRC issued Order EA-12-051, *Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation*, on March 12, 2012, to all US nuclear plants with an operating license, construction permit, or combined construction and operating license.

NRC Order EA-12-051 states, in part, "All licensees ... shall have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred." To this end, all licensees must provide:

- A primary and back-up level instrument that will monitor water level from the normal level to the top of the used fuel rack in the pool;
- A display in an area accessible following a severe event; and
- Independent electrical power to each instrument channel and provide an alternate remote power connection capability.

NEI 12-02, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation", provides guidance for complying with NRC Order EA-12-051.

This document includes three EALs that reflect the availability of the enhanced spent fuel pool level instrumentation associated with NRC Order EA-12-051. These EALs are included within existing IC RA2, and new ICs RS2 and RG2. These EALs will be implemented when the enhanced spent fuel pool level instrumentation is available for use.

2 KEY TERMINOLOGY USED

There are several key terms that appear throughout the EAL methodology. These terms are introduced in this section to support understanding of subsequent material. As an aid to the reader, the following table is provided as an overview to illustrate the relationship of the terms to each other.

EMERGENCY CLASSIFICATION LEVEL			
UNUSUAL EVENT	ALERT	SAE	GE
↓	↓	↓	↓
Initiating Condition	Initiating Condition	Initiating Condition	Initiating Condition
↓	↓	↓	↓
Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis	Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis	Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis	Emergency Action Level (1) • Operating Mode Applicability • Notes • Basis
(1) - When making an emergency classification, the Emergency Director must consider all information having a bearing on the proper assessment of an Initiating Condition. This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes and the informing Basis information. In the Recognition Category F matrices, EALs are referred to as Fission Product Barrier Thresholds; the thresholds serve the same function as an EAL.			

2.1 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)

2.1.1 UNUSUAL EVENT (UE)

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.

Purpose: The purpose of this classification is to assure that the first step in future response has been carried out, to bring the operations staff to a state of readiness, and to provide systematic handling of unusual event information and decision-making.

2.1.2 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.

Purpose: The purpose of this classification is to assure that emergency personnel are readily available to respond if the situation becomes more serious or to perform confirmatory radiation monitoring if required, and provide offsite authorities current information on plant status and parameters.

2.1.3 SITE AREA EMERGENCY (SAE)

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.

Purpose: The purpose of the SITE AREA EMERGENCY declaration is to assure that emergency response centers are staffed, to assure that monitoring teams are dispatched, to assure that personnel required for evacuation of near-site areas are at duty stations if the situation becomes more serious, to provide consultation with offsite authorities, and to provide updates to the public through government authorities.

2.1.4 GENERAL EMERGENCY (GE)

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.

Purpose: The purpose of the GENERAL EMERGENCY declaration is to initiate predetermined protective actions for the public, to provide continuous assessment of information from the licensee and offsite organizational measurements, to initiate additional measures as indicated by actual or potential releases, to provide consultation with offsite authorities, and to provide updates for the public through government authorities.

2.2 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.

Discussion: An IC describes an event or condition, the severity or consequences of which meets the definition of an emergency classification level. An IC can be expressed as a continuous, measurable parameter (e.g., RCS leakage), an event (e.g., an earthquake) or the status of one or more fission product barriers (e.g., loss of the RCS barrier).

Appendix 1 of NUREG-0654 does not contain example Emergency Action Levels (EALs) for each ECL, but rather Initiating Conditions (i.e., plant conditions that indicate that a radiological emergency, or events that could lead to a radiological emergency, has occurred). NUREG-0654 states that the Initiating Conditions form the basis for establishment by a licensee of the specific plant instrumentation readings (as applicable) which, if exceeded, would initiate the emergency classification. Thus, it is the specific instrument readings that would be the EALs.

Considerations for the assignment of a particular INITIATING CONDITION to an EMERGENCY CLASSIFICATION LEVEL are discussed in Section 3.

2.2.1 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.

Discussion: EAL statements may utilize a variety of criteria including instrument readings and status indications; observable events; results of calculations and analyses; entry into particular procedures; and the occurrence of natural phenomena.

2.2.2 FISSION PRODUCT BARRIER THRESHOLD

A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.

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

- Fuel Clad
- Reactor Coolant System (RCS)
- Containment

Upon determination that one or more fission product barrier thresholds have been exceeded, the combination of barrier loss and/or potential loss thresholds is compared to the fission product barrier IC/EAL criteria to determine the appropriate ECL.

In some accident sequences, the ICs and EALs presented in the Abnormal Radiation Levels/ Radiological Effluent (R) Recognition Category will be exceeded at the same time, or shortly after, the loss of one or more fission product barriers. This redundancy is intentional as the former ICs address radioactivity releases that result in certain offsite doses from whatever cause, including events that might not be fully encompassed by fission product barriers (e.g., spent fuel pool accidents, design containment leakage following a LOCA, etc.).

3 DESIGN OF THE STPEGS EMERGENCY CLASSIFICATION SCHEME

3.1 ASSIGNMENT OF EMERGENCY CLASSIFICATION LEVELS (ECLS)

An effective emergency classification scheme must incorporate a realistic and accurate assessment of risk, both to plant workers and the public. There are obvious health and safety risks in underestimating the potential or actual threat from an event or condition; however, there are also risks in overestimating the threat as well (e.g., harm that may occur during an evacuation). The emergency classification scheme attempts to strike an appropriate balance between reasonably anticipated event or condition consequences, potential accident trajectories, and risk avoidance or minimization.

There are a range of "non-emergency events" reported to the US Nuclear Regulatory Commission (NRC) staff in accordance with the requirements of 10 CFR § 50.72. Guidance concerning these reporting requirements, and example events, are provided in NUREG-1022. Certain events reportable under the provisions of 10 CFR § 50.72 may also require the declaration of an emergency.

In order to align each Initiating Conditions (IC) with the appropriate ECL, it was necessary to determine the attributes of each ECL. The goal of this process is to answer the question, "What events or conditions should be placed under each ECL?" The following sources provided information and context for the development of ECL attributes.

- Assessments of the effects and consequences of different types of events and conditions
- STPEGS abnormal and emergency operating procedure setpoints and transition criteria
- STPEGS Technical Specification limits and controls
- STPEGS Offsite Dose Calculation Manual (ODCM) radiological release limits
- Review of selected STPEGS Updated Final Safety Analysis Report (UFSAR) accident analyses
- Environmental Protection Agency (EPA) Protective Action Guidelines (PAGs)
- NUREG 0654, Appendix 1, Emergency Action Level Guidelines for Nuclear Power Plants
- Industry Operating Experience
- Input from subject matter experts at STPEGS

The following ECL attributes were created to aid in the development of ICs and Emergency Action Levels (EALs). The attributes may be useful in briefing and training settings (e.g., helping an Emergency Director understand why a particular condition is classified as an ALERT).

The attributes of each ECL are presented below.

3.1.1 UNUSUAL EVENT (UE)

An UNUSUAL EVENT, as defined in section 2.1.1, includes but is not limited to an event or condition that involves:

(A) A precursor to a more significant event or condition.
(B) A minor loss of control of radioactive materials or the ability to control radiation levels within the plant.
(C) A consequence otherwise significant enough to warrant notification to local, State and Federal authorities.

3.1.2 ALERT

An ALERT, as defined in section 2.1.2, includes but is not limited to an event or condition that involves:

(A) A loss or potential loss of either the fuel clad or Reactor Coolant System (RCS) fission product barrier.
(B) An event or condition that significantly reduces the margin to a loss or potential loss of the fuel clad or RCS fission product barrier.
(C) A significant loss of control of radioactive materials resulting in an inability to control radiation levels within the plant, or a release of radioactive materials to the environment that could result in doses greater than 1% of an EPA PAG at or beyond the site boundary.
(D) A HOSTILE ACTION occurring within the OWNER CONTROLLED AREA.

3.1.3 SITE AREA EMERGENCY (SAE)

A SITE AREA EMERGENCY, as defined in section 2.1.3, includes but is not limited to an event or condition that involves:

(A) A loss or potential loss of any two fission product barriers - fuel clad, RCS and/or containment.
(B) A precursor event or condition that may lead to the loss or potential loss of multiple fission product barriers within a relatively short period of time. Precursor events and conditions of this type include those that challenge the monitoring and/or control of multiple safety systems.
(C) A release of radioactive materials to the environment that could result in doses greater than 10% of an EPA PAG at or beyond the site boundary.
(D) A HOSTILE ACTION occurring within the plant PROTECTED AREA.

3.1.4 GENERAL EMERGENCY (GE)

A GENERAL EMERGENCY, as defined in section 2.1.4, includes but is not limited to an event or condition that involves:

(A) Loss of any two fission product barriers AND loss or potential loss of the third barrier - fuel clad, RCS and/or containment.

(B) A precursor event or condition that, unmitigated, may lead to a loss of all three fission product barriers. Precursor events and conditions of this type include those that lead directly to core damage and loss of containment integrity.

(C) A release of radioactive materials to the environment that could result in doses greater than an EPA PAG at or beyond the site boundary.

(D) A HOSTILE ACTION resulting in the loss of key safety functions (reactivity control, core cooling/RPV water level or RCS heat removal) or damage to spent fuel.

3.1.5 Risk-Informed Insights

Emergency preparedness is a defense-in-depth measure that is independent of the assessed risk from any particular accident sequence; however, the development of an effective emergency classification scheme can benefit from a review of risk-based assessment results. To that end, the development and assignment of certain ICs and EALs also considered insights from several site-specific probabilistic safety assessments (PSA - also known as probabilistic risk assessment, PRA). Some generic insights from this review included:

1. Accident sequences involving a prolonged loss of all AC power are significant contributors to core damage frequency. For this reason, a loss of all AC power for greater than 15 minutes, with the plant at or above Hot Shutdown, was assigned an ECL of SITE AREA EMERGENCY. Precursor events to a loss of all AC power were also included as an UNUSUAL EVENT and an ALERT.

   A station blackout coping analyses performed in response to 10 CFR § 50.63 and Regulatory Guide 1.155, *Station Blackout,* may be used to determine a time-based criterion to demarcate between a SITE AREA EMERGENCY and a GENERAL EMERGENCY. The time dimension is critical to a properly anticipatory emergency declaration since the goal is to maximize the time available for State and local officials to develop and implement offsite protective actions. 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.

2. For severe core damage events, uncertainties exist in phenomena important to accident progressions leading to containment failure. Because of these uncertainties, predicting the status of containment integrity may be difficult under severe accident conditions. This is why maintaining containment integrity alone following sequences leading to severe core damage is an insufficient basis for not escalating to a GENERAL EMERGENCY.

3. PSAs indicated that leading contributors to latent fatalities were sequences involving a containment bypass, a large Loss of Coolant Accident (LOCA) with early containment failure, a Station Blackout lasting longer than four hours, and a reactor coolant pump seal failure. The generic EAL methodology needs to be sufficiently rigorous to address these sequences in a timely fashion.

3.2 TYPES OF INITIATING CONDITIONS AND EMERGENCY ACTION LEVELS

The STPEGS methodology makes use of symptom-based, barrier-based and event-based ICs and EALs. Each type is discussed below.

Symptom-based ICs and EALs are parameters or conditions that are measurable over some range using plant instrumentation (e.g., core temperature, reactor coolant level, radiological effluent, etc.). When one or more of these parameters or conditions are off-normal, reactor operators will implement procedures to identify the probable cause(s) and take corrective action.

Fission product barrier-based ICs and EALs are the subset of symptom-based EALs that refer specifically to the level of challenge to the principal barriers against the release of radioactive material from the reactor core to the environment. These barriers are the fuel cladding, the reactor coolant system pressure boundary, and the containment. The barrier-based ICs and EALs consider the level of challenge to each individual barrier - potentially lost and lost - and the total number of barriers under challenge.

Event-based ICs and EALs define a variety of specific occurrences that have potential or actual safety significance. These include the failure of an automatic reactor scram/trip to shut down the reactor, natural phenomena (e.g., an earthquake), or man-made hazards such as a toxic gas release.

3.3 STPEGS DESIGN CONSIDERATIONS

The South Texas Project Electrical Generating Station (STPEGS) is composed of two units, each having an identical pressurized water reactor (PWR) Nuclear Steam Supply System (NSSS) and turbine generator (TG).

The NSSS is a Westinghouse Electric Corporation four-loop PWR. High-pressure light water serves as the coolant, neutron moderator, reflector, and solvent for the neutron absorber. The Reactor Coolant System (RCS), comprised of four parallel loops (each with a RCP and a steam generator [SG]), is used to transfer the heat generated in the core to the SGs using RCPs to circulate the water. RCS pressure is maintained by means of a pressurizer attached to the hot leg of one of the loops. The RCS is designed to circulate borated demineralized water at temperatures, pressures and flow rates consistent with the design thermal and hydraulic performance of the NSSS.

The Reactor Coolant Pressure Boundary Leak Detection System consists of temperature, level, humidity, and radioactivity sensors with associated instrumentation and alarms. Small leaks are detected by temperature and level changes of systems, increasing sump levels, and humidity and radioactivity concentration changes inside the Containment. Large leaks are detected by changes in reactor coolant inventory, changes in flow rates in process lines and changes in sump level.

Emergency Core Cooling System consists of three independent trains, each one capable of providing 100 percent of the required flow to the core in the unlikely event of a LOCA. Each train consists of one high-head safety injection pump and one low-head safety injection pump. Heat is removed from the system during recirculation by the residual heat removal heat exchanger (low-head pump only). The piping and valving associated with each of the three subsystems are identical. In the event of a steam pipe rupture, the ECCS provides adequate shutdown capability.

The Reactor Containment is a post-tensioned concrete cylinder with a steel liner plate, hemispherical top, and flat bottom. This structure provides a virtually leaktight barrier to prevent escape of fission products to the environment in the unlikely event of a loss of coolant accident (LOCA).

3.4 ORGANIZATION AND PRESENTATION OF GENERIC INFORMATION

The scheme's generic information is organized by Recognition Category in the following order.

- R- Abnormal Radiation Levels / Radiological Effluent – Section 6
- C - Cold Shutdown / Refueling System Malfunction – Section 7
- E - Independent Spent Fuel Storage Installation (ISFSI) – Section 8
- F - Fission Product Barrier – Section 9
- H - Hazards and Other Conditions Affecting Plant Safety – Section 10
- S - System Malfunction – Section 11

Each Recognition Category section contains a matrix showing the ICs and their associated EMERGENCY CLASSIFICATION LEVELS. The following information and guidance is provided for each IC:

- ECL – the assigned emergency classification level for the IC.
- Initiating Condition – provides a summary description of the emergency event or condition.
- Operating Mode Applicability – Lists the modes during which the IC and associated EAL(s) are applicable (i.e., are to be used to classify events or conditions).
- Emergency Action Level(s) – Provides indications that are considered to meet the intent of the IC.

For Recognition Category F, the fission product barrier thresholds are presented in tables and arranged by fission product barrier and the degree of barrier challenge (i.e., potential loss or loss). This presentation method shows the synergism among the thresholds, and supports accurate assessments.

Basis – Provides background information that explains the intent and application of the IC and EALs. In some cases, the basis also includes relevant source information and references.

3.5 IC AND EAL MODE APPLICABILITY

The STPEGS emergency classification scheme was developed recognizing that the applicability of ICs and EALs will vary with plant mode. For example, some symptom-based ICs and EALs can be assessed only during the power operations, startup, or hot standby/shutdown modes of operation when all fission product barriers are in place, and plant instrumentation and safety systems are fully operational. In the cold shutdown and refueling modes, different symptom-based ICs and EALs will come into play to reflect the opening of systems for routine maintenance, the unavailability of some safety system components and the use of alternate instrumentation.

The following table shows which Recognition Categories are applicable in each plant mode. The ICs and EALs for a given Recognition Category are applicable in the indicated modes.

MODE OF APPLICABILITY MATRIX

	Recognition Category					
Mode	**R**	**C**	**E**	**F**	**H**	**S**
Power Operations	X		X	X	X	X
Startup	X		X	X	X	X
Hot Standby	X		X	X	X	X
Hot Shutdown	X		X	X	X	X
Cold Shutdown	X	X	X		X	
Refueling	X	X	X		X	
Defueled	X	X	X		X	

STPEGS Operating Modes

Mode	Description	Criteria (Rx Power excludes decay heat)
1	Power Operations	Reactor Power > 5%, Keff $\geq$ 0.99 T Avg $\geq$ 350°F
2	Startup	Reactor Power $\leq$ 5%, Keff $\geq$ 0.99 T Avg $\geq$ 350°F
3	Hot Standby	Reactor Power 0% Keff < 0.99 T Avg $\geq$ 350°F
4	Hot Shutdown	Reactor Power 0% Keff < 0.99 350°F > T Avg > 200°F
5	Cold Shutdown	Reactor Power 0% Keff < 0.99 T Avg $\leq$ 200°F
6	Refueling	Reactor Power 0% Keff $\leq$ 0.95 T Avg $\leq$ 140°F Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.
	Defueled	All fuel removed from the reactor vessel (i.e., full core offload during refuel or extended outage)

4 STPEGS SCHEME DEVELOPMENT

4.1 GENERAL DEVELOPMENT PROCESS

The STPEGS ICs and EALs were developed to be unambiguous and readily assessable because both serve specific purposes. The IC is the fundamental event or condition requiring a declaration. The EAL(s) is the pre-determined threshold that defines when the IC is met. To this end, the STPEGS ICs and EALs were developed with input from key stakeholders such as Operations, Training, Health Physics, and Engineering. STPEGS specific indications, parameters and values were consistent with licensing basis documents, plant procedures, training, calculations, and drawings

Useful acronyms and abbreviations associated with the STPEGS emergency classification scheme are presented in Appendix A, Acronyms and Abbreviations. Those specific to STPEGS were included to be consistent with site terminology, site procedure, and training.

Many words or terms used in the STPEGS emergency classification scheme have scheme-specific definitions. These words and terms are identified by being set in all capital letters (i.e., ALL CAPS). The definitions are presented in Appendix B, Definitions.

4.2 CRITICAL CHARACTERISTICS

When crafting the scheme, STPEGS ensured that certain critical characteristics were met. These critical characteristics are listed below.

- The ICs, EALs, Operating Mode Applicability criteria, Notes and Basis information are consistent with industry guidance; while the actual wording may be different from NEI 99-01 Revision 6, the classification intent is maintained. With respect to Recognition Category F, the STPEGS scheme included a user-aid to facilitate timely and accurate classification of fission product barrier losses and/or potential losses. The user-aid logic is consistent with the classification logic presented in Section 9.
- EAL statements use objective criteria and observable values.
- ICs, EALs, Operating Mode Applicability and Note statements and formatting consider human factors and are user-friendly.
- The scheme facilitates upgrading and downgrading of the emergency classification where necessary.
- The scheme facilitates classification of multiple concurrent events or conditions.

4.3 INSTRUMENTATION USED FOR EALS

STPEGS incorporated instrumentation that is reliable and routinely maintained in accordance with site programs and procedures. Alarms referenced in EAL statements are those that are the most operationally significant for the described event or condition. EAL setpoints are within the calibrated range of the referenced instrumentation, and consider any automatic instrumentation functions that may impact accurate EAL assessment. In addition, EAL setpoint values do not use terms such as "off-scale low" or "off-scale high" since that type of reading may not be readily differentiated from an instrument failure. If instrumentation failures occur that have EALs associated with them (i.e., process radiation monitors) compensatory means of implementation may be used as described in plant procedures.

4.4 REFERENCES TO STPEGS AOPS AND EOPS

Some of the criteria/values used in several EALs and fission product barrier thresholds were drawn from STPEGS AOPs and EOPs. This approach was intended to maintain good alignment between operational diagnoses and emergency classification assessments. STPEGS verified the appropriate administrative controls are in place to ensure that a subsequent change to an AOP or EOP is screened to determine if an evaluation pursuant to 10 CFR 50.54(q) is required.

5 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

5.1 GENERAL CONSIDERATIONS

When making an emergency classification, the Emergency Director must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes and the informing Basis information. In the Recognition Category F matrices, EALs are referred to as Fission Product Barrier Thresholds; the thresholds serve the same function as an EAL.

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, Interim Staff Guidance, *Emergency Planning for Nuclear Power Plants.*

All emergency classification assessments should be based upon VALID indications, reports or conditions. A VALID indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy. For example, validation could be accomplished through an instrument channel check, response on related or redundant indicators, or direct observation by plant personnel. The validation of indications should be completed in a manner that supports timely emergency declaration.

For ICs and EALs that have a stipulated time duration (e.g., 15 minutes, 30 minutes, etc.), the Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time. If an ongoing radiological release is detected and the release start time is unknown, it should be assumed that the release duration specified in the IC/EAL has been exceeded, absent data to the contrary.

A planned work activity that results in an expected event or condition which meets or exceeds an EAL does not warrant an emergency declaration provided that 1) the activity proceeds as planned and 2) the plant remains within the limits imposed by the operating license. Such activities include planned work to test, manipulate, repair, maintain or modify a system or component. In these cases, the controls associated with the planning, preparation and execution of the work will ensure that compliance is maintained with all aspects of the operating

license provided that the activity proceeds and concludes as expected. Events or conditions of this type may be subject to the reporting requirements of 10 § CFR 50.72.

The assessment of some EALs is based on the results of analyses that are necessary to ascertain whether a specific EAL threshold has been exceeded (e.g., dose assessments, chemistry sampling, RCS leak rate calculation, etc.); the EAL and/or the associated basis discussion will identify the necessary analysis. In these cases, the 15-minute declaration period starts with the availability of the analysis results that show the threshold to be exceeded (i.e., this is the time that the EAL information is first available). The NRC expects licensees to establish the capability to initiate and complete EAL-related analyses within a reasonable period of time (e.g., maintain the necessary expertise on-shift).

While the EALs have been developed to address a full spectrum of possible events and conditions which may warrant emergency classification, a provision for classification based on operator/management experience and judgment is still necessary. This scheme provides the Emergency Director with the ability to classify events and conditions based upon judgment using EALs that are consistent with the Emergency Classification Level (ECL) definitions (refer to Category H). The Emergency Director will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated into the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

5.2 CLASSIFICATION METHODOLOGY

To make an emergency classification, the user will compare an event or condition (i.e., the relevant plant indications and reports) to an EAL(s) and determine if the EAL has been met or exceeded. The evaluation of an EAL(s) must be consistent with the related Operating Mode Applicability and Notes. If an EAL has been met or exceeded, then the IC is considered met and the associated ECL is declared in accordance with plant procedures.

When assessing an EAL that specifies a time duration for the off-normal condition, the "clock" for the EAL time duration runs concurrently with the emergency classification process "clock." For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01.

5.3 CLASSIFICATION OF MULTIPLE EVENTS AND CONDITIONS

When multiple emergency events or conditions are present, the user will identify all met or exceeded EALs. The highest applicable ECL identified during this review is declared. For example:

- If an ALERT EAL and a SITE AREA EMERGENCY EAL are met, whether at one unit or at two different units, a SITE AREA EMERGENCY should be declared.

There is no "additive" effect from multiple EALs meeting the same ECL. For example:

- If two ALERT EALs are met, whether at one unit or at two different units, an ALERT should be declared.

Related guidance concerning classification of rapidly escalating events or conditions is provided in Regulatory Issue Summary (RIS) 2007-02, *Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events*.

5.4 CONSIDERATION OF MODE CHANGES DURING CLASSIFICATION

The mode in effect at the time that an event or condition occurred, and prior to any plant or operator response, is the mode that determines whether or not an IC is applicable. If an event or condition occurs, and results in a mode change before the emergency is declared, the emergency classification level is still based on the mode that existed at the time that the event or condition was initiated (and not when it was declared). Once a different mode is reached, any new event or condition, not related to the original event or condition, requiring emergency classification should be evaluated against the ICs and EALs applicable to the operating mode at the time of the new event or condition.

For events that occur in Cold Shutdown or Refueling, escalation is via EALs that are applicable in the Cold Shutdown or Refueling modes, even if Hot Shutdown (or a higher mode) is entered during the subsequent plant response. In particular, the fission product barrier EALs are applicable only to events that initiate in the Hot Shutdown mode or higher.

5.5 CLASSIFICATION OF IMMINENT CONDITIONS

Although EALs provide specific thresholds, the Emergency Director must remain alert to events or conditions that could lead to meeting or exceeding an EAL within a relatively short period of time (i.e., a change in the ECL is IMMINENT). If, in the judgment of the Emergency Director, meeting an EAL is IMMINENT, the emergency classification should be made as if the EAL has been met. While applicable to all EMERGENCY CLASSIFICATION LEVELS, this approach is particularly important at the higher EMERGENCY CLASSIFICATION LEVELS since it provides additional time for implementation of protective measures.

5.6 EMERGENCY CLASSIFICATION LEVEL UPGRADING AND DOWNGRADING

An ECL may be downgraded when the event or condition that meets the highest IC and EAL no longer exists, and other site-specific downgrading requirements are met. If downgrading the ECL is deemed appropriate, the new ECL would then be based on a lower applicable IC(s) and EAL(s). The ECL may also simply be terminated.

The following approach to downgrading or terminating an ECL is recommended.

ECL	Action When Condition No Longer Exists
UNUSUAL EVENT	Terminate the emergency in accordance with plant procedures.
ALERT	Downgrade or terminate the emergency in accordance with plant procedures.
SITE AREA EMERGENCY with no long-term plant damage	Downgrade or terminate the emergency in accordance with plant procedures.
SITE AREA EMERGENCY with long-term plant damage	Terminate the emergency and enter recovery in accordance with plant procedures.
GENERAL EMERGENCY	Terminate the emergency and enter recovery in accordance with plant procedures.

As noted above, guidance concerning classification of rapidly escalating events or conditions is provided in RIS 2007-02.

5.7 CLASSIFICATION OF SHORT-LIVED EVENTS

As discussed in Section 3.2, event-based ICs and EALs define a variety of specific occurrences that have potential or actual safety significance. By their nature, some of these events may be short-lived and, thus, over before the emergency classification assessment can be completed. If an event occurs that meets or exceeds an EAL, the associated ECL must be declared regardless of its continued presence at the time of declaration. Examples of such events include a failure of the reactor protection system to automatically scram/trip the reactor followed by a successful manual scram/trip or an earthquake.

5.8 CLASSIFICATION OF TRANSIENT CONDITIONS

Many of the ICs and/or EALs contained in this document employ time-based criteria. These criteria will require that the IC/EAL conditions be present for a defined period of time before an emergency declaration is warranted. In cases where no time-based criterion is specified, it is recognized that some transient conditions may cause an EAL to be met for a brief period of time (e.g., a few seconds to a few minutes). The following guidance should be applied to the classification of these conditions.

<u>EAL momentarily met during expected plant response</u> - In instances where an EAL is briefly met during an expected (normal) plant response, an emergency declaration is not warranted provided that associated systems and components are operating as expected, and operator actions are performed in accordance with procedures.

<u>EAL momentarily met but the condition is corrected prior to an emergency declaration</u> – If an operator takes prompt manual action to address a condition, and the action is successful in correcting the condition prior to the emergency declaration, then the applicable EAL is not considered met and the associated emergency declaration is not required. For illustrative purposes, consider the following example.

An ATWS occurs and the auxiliary feedwater system fails to automatically start. Steam generator levels rapidly lower and the plant enters an inadequate RCS heat removal condition (a potential loss of both the fuel clad and RCS barriers). If an operator manually starts the auxiliary feedwater system in accordance with an EOP step and clears the inadequate RCS heat removal condition prior to an emergency declaration, then the classification should be based on the ATWS only.

It is important to stress that the 15-minute emergency classification assessment period is not a "grace period" during which a classification may be delayed to allow the performance of a corrective action that would obviate the need to classify the event; emergency classification assessments must be deliberate and timely, with no undue delays. The provision discussed above addresses only those rapidly evolving situations where an operator is able to take a successful corrective action prior to the Emergency Director completing the review and steps necessary to make the emergency declaration. This provision is included to ensure that any public protective actions resulting from the emergency classification are truly warranted by the plant conditions.

5.9 AFTER-THE-FACT DISCOVERY OF AN EMERGENCY EVENT OR CONDITION

In some cases, an EAL may be met but the emergency classification was not made at the time of the event or condition. This situation can occur when personnel discover that an event or condition existed which met an EAL, but no emergency was declared, and the event or condition no longer exists at the time of discovery. This may be due to the event or condition not being recognized at the time or an error that was made in the emergency classification process.

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR § 50.72 within one hour of the discovery of the undeclared event or condition. The licensee should also notify appropriate State and local agencies in accordance with the agreed upon arrangements.

5.10 RETRACTION OF AN EMERGENCY DECLARATION

Guidance on the retraction of an emergency declaration reported to the NRC is discussed in NUREG-1022.

6 ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

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

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
RU1 Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer. *Op. Modes: All*	**RA1** Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem THYROID CDE. *Op. Modes: All*	**RS1** Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem THYROID CDE. *Op. Modes: All*	**RG1** Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem THYROID CDE. *Op. Modes: All*
RU2 UNPLANNED loss of water level above irradiated fuel. *Op. Modes: All*	**RA2** Significant lowering of water level above, or damage to, irradiated fuel. *Op. Modes: All*	**RS2** Spent fuel pool level at 40'-4" or lower. *Op. Modes: All*	**RG2** Spent fuel pool level cannot be restored to at least 40'-4" for 60 minutes or longer. *Op. Modes: All*
	RA3 Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown. *Op. Modes: All*		

RU1

ECL: UNUSUAL EVENT

Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer.

Operating Mode Applicability: **ALL**

Emergency Action Levels: (1 or 2 or 3)

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** of the following radiation monitor greater than the values listed in Table R1 column "UE" for 60 minutes or longer:

Table R1: Effluent Monitors					
Release Point	Monitor	GE	SAE	ALERT	UE
Unit Vent	RT-8010B	1.50 E+08 µCi/sec	1.50 E+07 µCi/sec	1.50 E+06 µCi/sec	1.40 E+05 µCi/sec
Main Steam Lines	RT-8046 thru 8049	4.00 E+02 µCi/cm^3	4.00 E+01 µCi/cm^3	4.00 E+00 µCi/cm^3	5.00 E-02 µCi/cm^3

(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.

(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.

Basis:

This IC addresses a potential lowering 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.

STPEGS incorporated 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 normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways.

EAL #2- 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 #3- 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 RA1.

RU1: EAL-1 Selection Basis

The Unit Vent and Main Steam Line monitor readings were included in this EAL because they give instantaneous indications of a monitored gaseous release exceeding twice the ODCM limits. Normal gaseous effluents are due to planned RCB purges and monitored by the Unit Vent. The Main Steam Line monitor readings were included because they correspond to a concentration that would result in a release rate of twice the ODCM limits if there were a release via the Power Operated Relief Valves (PORVs) or Safety Relief Valves. A release from the PORVs or Safety Relief Valves is not a normal effluent pathway but meets the intent of the EAL.

The Unit Vent and Main Steam Line release values are based on Calculation No. STPNOC013-CALC-002, Rev. 2.

RU1: EAL-2, 3 Selection Basis

For EAL-2, there are two effluent radiation monitors, RT-8038 (liquid) and RT-8010B (gaseous), however only RT-8010B was included. The alarm setpoint for the gaseous effluent radiation monitor RT-8010B is set at the ODCM limits. An indication of two times the alarm setpoint (two times the ODCM limit) would allow operators time to secure the release prior to meeting this EAL. The liquid effluent radiation monitor RT-8038 was not included in EAL-2 because the activity in liquid discharges is normally the several orders of magnitude lower than the ODCM limits. In order to alert personnel to significant changes in the liquid effluent activity, the alarm setpoint for RT-8038 is normally set several orders of magnitude below the ODCM limits. Setting the alarm setpoint for RT-8038 at the ODCM limit would remove this capability and violate the intent of the EAL.

For EAL-3, sample analysis could be used as a backup for the effluent monitor indications.

REFERENCES:

1. Calculation No: STPNOC013-CALC-002 Rev. 2, Radiological Release Thresholds for Emergency Action Levels
2. Offsite Dose Calculation Manual (ODCM), Rev. 17, Part B3.0 to B4.9
3. UFSAR, Rev. 14, Section 11.5.2.3.3 and 11.5.2.5.3 (monitor descriptions)
4. UFSAR, Rev. 14, Section 11.5.2.4.4 (liquid waste processing monitor)

RU2

ECL: UNUSUAL EVENT

Initiating Condition: UNPLANNED loss of water level above irradiated fuel.

Operating Mode Applicability: ALL

Emergency Action Level:

(1) a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following:

- Visual Observation

  OR

- Annunciator alarm on lampbox 22M02 Window F-5 "SFP WATER LVL HI/LO"

  OR

- Spent fuel in the ICSA **AND** Annunciator alarm on lampbox 22M02 Window F-6 "SFP Trouble" **AND** Plant Computer point FCLC1420 "REFLNG CAV LVL IN CNTMT" (ICSA Water Level HI/LO) is in alarm

AND

b. UNPLANNED rise in area radiation levels on ANY of the following radiation monitors.

- RE-8055 (68' RCB) - Mode 5 or 6 only

  OR

- RE-8099 (68' RCB) - Mode 5 or 6 only

  OR

- RE-8090 (68' FHB)

Basis:

This IC addresses a lowering in water level above irradiated fuel sufficient to cause elevated radiation levels. This condition could be a precursor to a more serious event and is also indicative of a minor loss in the ability to control radiation levels within the plant. It is therefore a potential degradation in the level of safety of the plant.

A water level lowering will be primarily determined by indications from available level instrumentation. Other sources of level indications may include reports from plant personnel (e.g., from a refueling crew) or video camera observations. A significant drop in the water level may also cause a rise in the radiation levels of adjacent areas that can be detected by monitors in those locations.

The effects of planned evolutions should be considered. For example, a refueling bridge area radiation monitor reading may rise due to planned evolutions such as lifting of the reactor vessel head or movement of a fuel assembly. Note that this EAL is applicable only in cases where the elevated reading is due to an UNPLANNED loss of water level.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC RA2.

RU2: EAL-1 Selection Basis

Hi/Lo level sensors are located in the Spent Fuel Pool (LSHL 1401) and the RCB, In Containment Storage Area (ICSA) (LSHL 1420). If level in the Spent Fuel Pool rises or lowers by more than 6 inches above or below the normal water level of 66'-6" (UFSAR 9.1.2.1), the "SFP WATER LEVEL HI/LO" lampbox 22M02 window F-5 annunciator alarm is received in the Control Room (0POP09-AN-22M2, Annunciator Lampbox 22M02 Response Instructions).

Although the ICSA has a Hi/LO level sensor, there is not an annunciator in the Control Room similar to the one for the Spent Fuel Pool. There is however, a "SFP TROUBLE" lampbox 22M02 window F-6 annunciator in the control room. One of the inputs to this alarm is FC-LSHL-1420, the ICSA HI/LO level sensor. Since no fuel is located in the ICSA in modes 1-4, this EAL only applies in modes 5 or 6.

Area radiation monitors RE-8055 and RE-8099 are located are located in the RCB 68' elevation on the bioshield wall close to the refueling cavity. Area radiation monitor RE-8090 is located in the Fuel Handling Building on 68' Elevation near the Spent Fuel Pool.

Expected radiation levels for a loss of water level can range from a few mR/hr to thousands of R/hr.
For a drop of water level of approximately 14' (from 66'-6" to 51'-10") with approximately 13' of water over the top of any array, the dose rate would be expected not to exceed 2.5 mR/hr, above background. This assumes 42 hours of decay with a full core off-load (section 9 of STP UFSAR).

For a significant drop of water level that would still cover the arrays, the radiation levels could range from several hundred R/hr to over a thousand R/hr on and around the 68' elevation deck (table C-5 NUREG CR/0649).

REFERENCES:

1. 0POP09-AN-22M2, Rev. 25, Annunciator Lampbox 22M02 Response Instructions F-5 and F-6 Window (level alarms)
2. 0POP04-FC-0001, Rev. 29, Loss of Spent Fuel Pool Level or Cooling (level alarms)
3. Technical Specification, amendment 104 (Unit 1) and 91 (Unit 2), Section 5.6.2 (Design water level)
4. UFSAR, Rev. 16, Section 9.1.2.1 (Dose rates)
5. UFSAR, Rev. 16, Section 9.1.2.2 (Normal water level)
6. NUREG CR/0649 (Dose rates), reference only (not included in submittal)
7. Drawing 5R219F05028#1 Spent Fuel Pool Cooling and Cleanup System (level sensors)
8. UFSAR, Rev. 15, table 12.3.4-1, Area Radiation Monitors

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: ALL

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 values listed in Table R1 column "ALERT" for 15 minutes or longer:

Table R1: Effluent Monitors					
Release Point	**Monitor**	**GE**	**SAE**	**ALERT**	**UE**
Unit Vent	RT-8010B	1.50 E+08 µCi/sec	1.50 E+07 µCi/sec	1.50 E+06 µCi/sec	1.40 E+05 µCi/sec
Main Steam Lines	RT-8046 thru 8049	4.00 E+02 µCi/cm³	4.00 E+01 µCi/cm³	4.00 E+00 µCi/cm³	5.00 E-02 µCi/cm³

(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem THYROID CDE at or beyond the SITE BOUNDARY.

(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.

(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.

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.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC RS1.

RA1: EAL-1 Selection Basis

The Unit Vent and Main Steam Line monitor readings were included in this EAL because they give instantaneous indications of a monitored gaseous release meeting the EAL threshold values of 10 mrem TEDE or 50 mrem CDE THYROID at the SITE BOUNDARY. Gaseous releases from the plant are monitored by the Unit Vent. The Main Steam Line monitor readings correspond to a concentration that would result in a release rate meeting the EAL threshold values if there were a release via the Power Operated Relief Valves (PORVs) or Safety Relief Valves.

The Unit Vent and Main Steam Line release values are based on Calculation No. STPNOC013-CALC-002, Rev. 2. The adjusted values used in this EAL were conservatively truncated by less than 1% of the calculated values to ensure they are readily assessable.

RA1: EAL-2, 3, 4 Selection Basis

N/A

REFERENCES:

1. Calculation No: STPNOC013-CALC-002 Rev. 2., Radiological Release Thresholds for Emergency Action Levels
2. UFSAR, Rev. 14, Section 11.5.2.3.3 and 11.5.2.5.3 (monitor descriptions)
3. UFSAR, Rev. 14, 11.5.2.4.4 (liquid waste processing monitor)

ECL: ALERT

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel.

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2 or 3)

(1) Uncovery of irradiated fuel in the REFUELING PATHWAY.

(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:

- FHB Exhaust, RT-8035 or RT-8036 greater than 1.00 E-1 $\mu Ci/cm^3$

  OR

- ARM (68' FHB), RE-8090 greater than 1,500 mR/hr

OR

b. Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by **ANY** of the following RCB radiation monitor readings (Mode 5 or 6 only).

- ARMs (68' RCB), RE-8055 or RE-8099 greater than 850 mR/hr.

NOTE
EAL-3 is not applicable until the enhanced SFP level instrumentation is available for use.

(3) Lowering of spent fuel pool level to 49'-10" or lower.

Basis:

This IC addresses events that have caused IMMINENT or actual damage to an irradiated fuel assembly, *or a significant lowering of water level within the spent fuel pool or Inside Containment Storage Area (ICSA).*These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC E-HU1.

Escalation of the emergency would be based on either Recognition Category R or C ICs.

EAL #1- This EAL escalates from RU2 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used (e.g., a boil-off curve). Classification of an event using this EAL should be based on the totality of available indications, reports and observations. While an area radiation monitor could detect a rise in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.
A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

EAL #2- This EAL addresses a release of radioactive material caused by mechanical damage to irradiated fuel. Damaging events may include the dropping, bumping or binding of an assembly, or dropping a heavy load onto an assembly. A rise in readings on radiation monitors should be considered in conjunction with in-plant reports or observations of a potential fuel damaging event (e.g., a fuel handling accident).

EAL #3- Spent fuel pool water level at this value is within the lower end of the level range necessary to prevent significant dose consequences from direct gamma radiation to personnel performing operations in the vicinity of the spent fuel pool. This condition reflects a significant loss of spent fuel pool water inventory and thus it is also a precursor to a loss of the ability to adequately cool the irradiated fuel assembles stored in the pool.

Escalation of the EMERGENCY CLASSISFICATION LEVEL would be via ICs RS1 or RS2.

RA2: EAL-2 Selection Basis:

The calculated airborne source term and radiation monitor responses for a fuel handling accident in the FHB is based on Calculation STPNOC013-CALC-005 Rev.2. The threshold value of 1500 mR/hr for area radiation monitor RE-8090 was truncated less than 4% from the calculated value to ensure the threshold was readily assessable. Threshold values for FHB Exhaust Monitors RT-8035 and RT-8036 were also included because they are accident monitors that are sensitive to noble gases which are expected to be present if irradiated fuel is damaged. The calculated monitor reading for RT-8035 and RT-8036 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.

The calculated airborne source term and radiation monitor response for a fuel handling accident in the RCB is based on Calculation STPNOC013-CALC-005 Rev.2. The threshold value of 850 mR/hr for RE-8055 and RE-8099 was truncated less than 2% from the calculated value to ensure the threshold is readily assessable.

RA2: EAL-3 Selection Basis:

Spent Fuel Pool level of 49'- 10" (Level 2) is a site specific level based on the guidance provided in NEI 12-02, Revision 1, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licensees with Regard to Reliable Spent Fuel Pool Instrumentation", August 2012.

In NRC Order EA-12-051 and NEI 12-02, Level 2 is defined as the "*level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck ...* "

The STP UFSAR identifies the top of the Spent Fuel Storage Racks at 39'- 10". The guidance in NEI 12-02 indicates that 10' of water above the top of the Spent Fuel Storage Racks provides substantial radiation shielding. Ten feet of water above the Spent Fuel Storage Racks is 49'- 10", the threshold value for this EAL.

Reference 6 identifies the site specific levels of the proposed SFP level instrument and identifies the Level 2 criteria as 49'– 10".

REFERENCES:

1. Calculation No.: STPNOC013-CALC-005 Rev.2, Fuel Handling Accident Monitor Response for Emergency Action Levels.
2. UFSAR, Rev. 16, Section 9.1.2.1 (SFP Rad levels)
3. UFSAR, Rev. 16, Section 9.1.2.2 (SFP top of Racks)
4. NRC Order EA-12-051 (SFP levels)
5. NEI 12-02, Rev. 1 (SFP levels)
6. South Texas Project (STP) Overall Integrated Plan for Implementation of Unit 1 & Unit 2 Spent Fuel Pool Level Instrumentation to Meet NRC Order EA-12-051, Rev. 0, NOC –AE-13002959

RA3

ECL: ALERT

Initiating Condition: Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2)

Note: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

(1) Dose rate greater than 15 mR/hr in **ANY** of the following areas:

- Control Room ARM (RE-8066)

OR

- Central Alarm Station (CAS) by radiation survey

(2) An UNPLANNED event results in radiation levels that prohibit or impede access to **ANY** of the areas listed in Table H3/R2:

TABLE H3/R2: Plant Areas Requiring Access		
MODE 4	RCB	RHR Heat Exchanger Rooms
	MAB	51 ft Room 335
	EAB	Roof, MCC 1G8, 4.16KV Switchgear Rooms
MODE 5	EAB	4.16KV Switchgear Rooms

Basis:

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Emergency Director should consider the cause of the higher radiation levels and determine if another IC may be applicable.

For EAL #2, an ALERT declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the higher radiation levels. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., installing temporary shielding, requiring use of non-routine protective equipment, requesting an extension in dose limits beyond normal administrative limits).

An emergency declaration is not warranted if any of the following conditions apply.

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the elevated radiation levels). For example, the plant is in Mode 1 when the radiation rise occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The higher radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via Recognition Category R, C or F ICs.

RA3: EAL-1, EAL-2 Selection Basis:

The NEI 99-01 value of 15 mR/hr is derived from the GDC 19 value of 5 rem in 30 days with adjustment for expected occupancy times. The rooms listed in EAL-1 require continuous occupancy to maintain normal plant operation, or to perform a normal cooldown or shutdown.

The areas listed in EAL-2 apply to areas that contain equipment necessary for plant operations, cooldown, or shutdown. Assuming all plant equipment is operating as designed, Normal operations and safe shutdown equipment operation is capable from the Main Control Room (MCR). The plant is able to transition into a hot shutdown from the MCR, therefore H3/R2 is a list of plant rooms or areas with entry-related mode applicability that contain equipment which require a manual/local action necessary following entry into hot shutdown (establish Residual Heat Removal shutdown cooling, disable operation of charging and ECCS equipment, and limit dilution pathways) and subsequent entry into cold shutdown (disable operation of ECCS equipment). After achieving cold shutdown it is assumed that the plant will be maintained in a cold shutdown condition.

REFERENCES:

1. General Design Criteria 19
2. 0POP03-ZG-0008, Rev. 56, Power Operations
3. 0POP03-ZG-0006, Rev. 54, Plant Shutdown from 100% to Hot Standby
4. 0POP03-ZG-0007, Rev. 71, Plant Cooldown

RS1

ECL: SITE AREA EMERGENCY

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem THYROID CDE.

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2 or 3)

Notes:

- The Emergency Director should declare the SITE AREA EMERGENCY 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 values listed in Table R1 column "SAE" for 15 minutes or longer:

Table R1: Effluent Monitors

Release Point	Monitor	GE	SAE	ALERT	UE
Unit Vent	RT-8010B	1.50 E+08 μCi/sec	1.50 E+07 μCi/sec	1.50 E+06 μCi/sec	1.40 E+05 μCi/sec
Main Steam Lines	RT-8046 thru 8049	4.00 E+02 μCi/cm^3	4.00 E+01 μCi/cm^3	4.00 E+00 μCi/cm^3	5.00 E-02 μCi/cm^3

(2) Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem THYROID CDE at or beyond the SITE BOUNDARY.

(3) 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.

Basis:

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA PROTECTIVE ACTION GUIDES (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

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 10% of the EPA PAG of 1,000 mrem while the 500 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.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC RG1.

RS1: EAL-1 Selection Basis:

The Unit Vent and Main Steam Line monitor readings were included in this EAL because they give instantaneous indications of a monitored gaseous release meeting the EAL threshold values of 100 mrem TEDE or 500 mrem CDE THYROID at the SITE BOUNDARY. Gaseous releases from the plant are monitored by the Unit Vent. The Main Steam Line monitor readings correspond to a concentration that would result in a release rate meeting the EAL threshold values if there were a release via the Power Operated Relief Valves (PORVs) or Safety Relief Valves.

The Unit Vent and Main Steam Line release values are based on Calculation No. STPNOC013-CALC-002 Rev.2. The adjusted values used in this EAL were conservatively truncated by less than 1% of the calculated values to ensure they are readily assessable.

RS1: EAL-2, EAL-3 Selection Basis:

N/A

REFERENCES:

1. Calculation No: STPNOC013-CALC-002 Rev.2, Radiological Release Thresholds for Emergency Action Levels
2. UFSAR Section, Rev. 14, Section 11.5.2.3.3 and 11.5.2.5.3 (monitor descriptions)

RS2

ECL: SITE AREA EMERGENCY

Initiating Condition: Spent fuel pool level at 40'-4" or lower.

Operating Mode Applicability: ALL

Emergency Action Level:

NOTE
EAL-1 is not applicable until the enhanced SFP level instrumentation is available for use.

(1) Lowering of spent fuel pool level to 40'-4" or lower.

Basis:

This IC addresses a significant loss of spent fuel pool inventory control and makeup capability leading to IMMINENT fuel damage. This condition entails major failures of plant functions needed for protection of the public and thus warrant a SITE AREA EMERGENCY declaration.

It is recognized that this IC would likely not be met until well after another SITE AREA EMERGENCY IC was met; however, it is included to provide classification diversity.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC RG1 or RG2.

RS2: EAL-1 Selection Basis:

Spent Fuel Pool level of 40'- 4" (Level 3) is a site specific level based on the guidance provided in NEI 12-02, Revision 1, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation", August 2012.

In NRC Order EA-12-051 and NEI 12-02, Level 3 is defined as "*level where fuel remains covered and actions to implement make-up water addition should no longer be deferred*."

The STP UFSAR identifies the top of the Spent Fuel Storage Racks at 39'- 10".

Reference 4 identifies the site specific levels for the proposed SFP level instrumentation and identifies the Level 3 criteria as 40'– 4".

REFERENCES:

1. UFSAR, Rev. 16, Section 9.1.2.2 (SFP top of Racks)
2. NRC Order EA-12-051 (SFP Levels)
3. NEI 12-02, Revision 1, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation", August 2012
1. South Texas Project (STP) Overall Integrated Plan for Implementation of Unit 1 & Unit 2 Spent Fuel Pool Level Instrumentation to Meet NRC Order EA-12-051, Rev. 0, NOC –AE-13002959

ECL: GENERAL EMERGENCY

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem THYROID CDE.

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2 or 3)

Notes:

- The Emergency Director should declare the GENERAL EMERGENCY 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 values listed in Table R1 column "GE" for 15 minutes or longer:

Table R1: Effluent Monitors

Release Point	Monitor	GE	SAE	ALERT	UE
Unit Vent	RT-8010B	1.50 E+08 μCi/sec	1.50 E+07 μCi/sec	1.50 E+06 μCi/sec	1.40 E+05 μCi/sec
Main Steam Lines	RT-8046 thru 8049	4.00 E+02 μCi/cm^3	4.00 E+01 μCi/cm^3	4.00 E+00 μCi/cm^3	5.00 E-02 μCi/cm^3

(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.

(3) Field survey results indicate **EITHER** of the following at or the SITE BOUNDARY:

- Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer.

  OR

- Analyses of field survey samples indicate THYROID CDE greater than 5,000 mrem for one hour of inhalation.

Basis:

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA PROTECTIVE ACTION GUIDES (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

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 the EPA PAG of 1,000 mrem while the 5,000 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.

RG1: EAL-1 Selection Basis:

The Unit Vent and Main Steam Line monitor readings were included in this EAL because they give instantaneous indications of a monitored gaseous release meeting the EAL threshold values of 1000 mrem TEDE or 5000 mrem CDE THYROID at the SITE BOUNDARY. Gaseous releases from the plant are monitored by the Unit Vent. The Main Steam Line monitor readings correspond to a concentration that would result in a release rate meeting the EAL threshold values if the release was via the Power Operated Relief Valves (PORVs) or Safety Relief Valves.

The Unit Vent and Main Steam Line release values are based on Calculation No. STPNOC013-CALC-002 Rev.2. The adjusted values used in this EAL were conservatively truncated by less than 1% of the calculated values to ensure they are readily assessable.

RG1: EAL-2, EAL-3 Selection Basis:

N/A

REFERENCES:

1. Calculation No: STPNOC013-CALC-002 Rev.2, Radiological Release Thresholds for Emergency Action Levels,
2. STP UFSAR, Rev. 14, Section 11.5.2.3.3 and 11.5.2.5.3 (monitor descriptions)

ECL: GENERAL EMERGENCY

Initiating Condition: Spent fuel pool level cannot be restored to at least 40'-4" for 60 minutes or longer.

Operating Mode Applicability: ALL

Emergency Action Level:

Note: The Emergency Director should declare the GENERAL EMERGENCY promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.

NOTE
EAL-1 is not applicable until the enhanced SFP level instrumentation is available for use.

(1) Spent fuel pool level cannot be restored to at least40'-4" for 60 minutes or longer.

Basis:

This IC addresses a significant loss of spent fuel pool inventory control and makeup capability leading to a prolonged uncovery of spent fuel. This condition will lead to fuel damage and a radiological release to the environment.

It is recognized that this IC would likely not be met until well after another GENERAL EMERGENCY IC was met; however, it is included to provide classification diversity.

RG2: EAL-1 Selection Basis:

The Spent Fuel Pool level of 40'- 4" (Level 3) is a site specific level based on the guidance provided in NEI 12-02, Revision 1, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation", August 2012.

In NRC Order EA-12-051 and NEI 12-02, Level 3 is defined as "*level where fuel remains covered and actions to implement make-up water addition should no longer be deferred*."

The STP UFSAR identifies the top of the Spent Fuel Pool Racks at 39'- 10".

Reference 4 identifies the site specific levels of the proposed level instrumentation and identifies the Level 3 criteria as 40'- 4".

REFERENCES:

1. UFSAR, Rev. 16, Section 9.1.2.2 (SFP top of Racks)
2. NRC Order EA-12-051 (SFP Levels)
3. NEI 12-02, Rev. 1, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation", August 2012
4. South Texas Project (STP) Overall Integrated Plan for Implementation of Unit 1 & Unit 2 Spent Fuel Pool Level Instrumentation to Meet NRC Order EA-12-051, Rev. 0, NOC –AE-13002959

7 COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

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

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
CU1 UNPLANNED loss of RCS inventory for 15 minutes or longer. *Op. Modes:* 5,6	**CA1** Loss of RCS inventory. *Op. Modes:* 5,6	**CS1** Loss of RCS inventory affecting core decay heat removal capability. *Op. Modes:* 5,6	**CG1** Loss of RCS inventory affecting fuel clad integrity with containment challenged. *Op. Modes:* 5,6
CU2 Loss of **ALL** but one AC power source to emergency buses for 15 minutes or longer. *Op. Modes:* 5,6 *Defueled*	**CA2** Loss of **ALL** offsite and **ALL** onsite AC power to emergency buses for 15 minutes or longer. *Op. Modes:* 5,6 , *Defueled*		
CU3 UNPLANNED rise in RCS temperature. *Op. Modes:* 5,6	**CA3** Inability to maintain the plant in cold shutdown. *Op. Modes:* 5,6		
CU4 Loss of Vital DC power for 15 minutes or longer. *Op. Modes:* 5,6			
CU5 Loss of **ALL** onsite or offsite communications capabilities. *Op. Modes:* 5,6, *Defueled*			
	CA6 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. *Op. Modes:* 5,6		

CU1

ECL: UNUSUAL EVENT

Initiating Condition: UNPLANNED loss of RCS inventory for 15 minutes or longer.

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2)

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

(1) UNPLANNED loss of reactor coolant results in RCS level below the procedurally required limit for 15 minutes or longer.

(2) a. RCS level cannot be monitored.

AND

b. UNPLANNED rise in **ANY** of the following sump or tank levels in Table C2:

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

Basis:

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor RCS level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that lower RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an UNUSUAL EVENT due to the reduced water inventory that is available to keep the core covered.

EAL #1- recognizes that the minimum required RCS level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is specified in the applicable STP operating procedure.

The 15-minute threshold duration allows sufficient time for prompt operator actions to restore and maintain the expected water level. This criterion excludes transient conditions causing a brief lowering of water level.

EAL #2- addresses a condition where all means to determine RCS level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS

Continued loss of RCS inventory may result in escalation to the ALERT EMERGENCY CLASSIFICATION LEVEL via either IC CA1 or CA3.

CU1 – EAL-1 Selection Basis:

RCS inventory is maintained above the reactor vessel flange (39'-3") during refueling outages per 0POP03-ZG-0007, Plant Cooldown. RCS level may be lowered below the vessel flange for specific purposes (e.g., head removal, mid-loop operations) as described in 0POP03-ZG-0009, Mid-Loop Operation. The 15 minute time frame allows for prompt operator actions to restore RCS level in the event of an UNPLANNED lowering of RCS level below the prescribed operating limit.

CU1 – EAL-2 Selection Basis:

This EAL includes two conditions. The first condition is the inability to monitor RCS level and the second condition provides secondary indications that inventory loss may be occurring.

The secondary indicators of inventory loss include a list of tanks/sumps found in 0POP04-RC-0003, Excessive RCS Leakage. Since other system leaks could rise levels in various tanks and sumps, the list has been limited to the tanks and sumps that would have the highest probability of indicating RCS leakage inside the Reactor Containment Building.

Although procedure 0POP04-RC-0003 is designated for use in modes 1-4, its logic is applicable to this EAL.

REFERENCES:

1. 0POP04-RC-0003, Rev. 18, Excessive RCS Leakage
2. 0POP03-ZG-0007, Rev. 71, Plant Cooldown
3. 0POP03-ZG-0009, Rev. 59, Mid-Loop Operation

CU2

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: 5, 6, Defueled

Emergency Action Level:

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

(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.

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.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as an ALERT because of the additional time available to restore another power source to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an emergency bus. 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 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 fed from an onsite or offsite power source.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.
The subsequent loss of the remaining single power source would escalate the event to an ALERT in accordance with IC CA2.

CU2: EAL-1 Selection Criteria:

The condition indicated by this EAL is the degradation of the offsite and onsite power systems such that any additional single failure would results in a loss of all AC power. This condition is an UNUSUAL EVENT during modes 5, 6 and Defueled because of the additional time available to restore power due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. In modes 1-4, this condition is an ALERT as described in SA1.

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

CU3

ECL: UNUSUAL EVENT

Initiating Condition: UNPLANNED rise in RCS temperature.

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2)

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

(1) UNPLANNED rise in RCS temperature to greater than 200 °F (Tavg).

(2) Loss of **ALL** RCS temperature and RCS level indication for 15 minutes or longer.

__Basis:__

This IC addresses an UNPLANNED rise in RCS temperature above the Technical Specification cold shutdown temperature limit, or the inability to determine RCS temperature and level, represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Director should also refer to IC CA3.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

EAL #1- involves a loss of decay heat removal capability, or an addition of heat to the RCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

During an outage, the level in the reactor vessel will normally be maintained above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid rise in reactor coolant temperature depending on the time after shutdown.

EAL #2- reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication. Escalation to ALERT would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

CU3: EAL-1 Selection Basis:

An UNPLANNED temperature rise above 200 °F would result in an UNPLANNED mode change due to the inability to control RCS temperature. Mode 4 (Hot Shutdown) would be entered when Tavg exceeds 200 °F (Reference 1).

CU3: EAL-2 Selection Basis:

N/A

REFERENCES:

1. Technical Specifications Table 1.2 (Mode, Temperature, Power, k_{eff} Table)

CU4

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: 5, 6

Emergency Action Level:

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

(1) Indicated voltage is less than 105.5 VDC on required Vital DC buses for 15 minutes or longer.

Basis:

This IC addresses a loss of Vital DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or refueling mode. In these modes, the core decay heat load has been significantly reduced, and coolant system temperatures and pressures are lower; these conditions extend the time available to restore a vital DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, "required" means the Vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if Train A and C are out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an UNUSUAL EVENT. A loss of Vital DC power to Train A and/or C would not warrant an emergency classification.

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

Depending upon the event, escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC CA1 or CA3, or an IC in Recognition Category R.

CU4 – EAL-1 Selection Basis:

The minimum voltage for Class 1E 125 VDC battery buses was determined in calculation 13-DJ-006, Rev. 3 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. Calculation 13-DJ-006, Rev. 0, 125 VDC Battery Four Hour Coping Analysis
2. 0POP05-EO-EC00, Rev. 23, Loss of All AC Power

CU5

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: 5, 6, Defueled

Emergency Action Levels: (1 or 2 or 3)

(1) Loss of **ALL** of the following Onsite communication methods in Table C4.

(2) Loss of **ALL** of the following Offsite Response Organization (ORO) communication methods in Table C4.

(3) Loss of **ALL** of the following NRC communication methods in Table C4.

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

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 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.

CU5: EAL-1, EAL-2, and 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, Rev. 8, Emergency Communications

CA1

ECL: ALERT

Initiating Condition: Loss of RCS inventory.

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2)

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

(1) Loss of RCS inventory as indicated by level less than 32 ft. 9 inch (+ 6 inches above hot leg centerline).

(2) a. RCS level cannot be monitored for 15 minutes or longer

AND

b. UNPLANNED rise in **ANY** of the following sump or tank levels in Table C2 due to a loss of reactor vessel/RCS inventory.

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

Basis:

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

EAL #1- A lowering of water level below elevation 32'- 9" indicates that operator actions have not been successful in restoring and maintaining reactor vessel/ water level. The heat-up rate of the coolant will rise as the available water inventory is reduced. A continuing reduction in water level will lead to core uncovery. Although related, EAL #1 is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Residual Heat Removal suction point). Arise in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

EAL #2- The inability to monitor reactor vessel/RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to

ensure they are indicative of leakage from the reactor vessel/RCS. The 15-minute duration for the loss of level indication was chosen because it is half of the EAL duration specified in IC CS1

If the reactor vessel/RCS inventory level continues to lower, then escalation to SITE AREA EMERGENCY would be via IC CS1.

CA1: EAL-1 Selection Basis:

The minimum RCS level at which an RHR pump can be started per 0POP02-RH-0001 is 32 feet 9 inches (+ 6 inches above hot leg centerline). If RCS inventory is reduced below this level, normal decay heat removal systems may not be available for core cooling. This threshold is not applicable to reduced inventory vacuum fill since this is a controlled evolution and not indicative of an RCS loss.

CA1: EAL-2 Selection Basis:

The tanks/sumps selected for this EAL were obtained from 0POP04-RC-0003, Excessive RCS Leakage. Since other system leaks could raise levels in various tanks and sumps, the list was limited to the tanks and sumps that would have the highest probability of indicating RCS leakage inside the Reactor Containment Building.

Although procedure 0POP04-RC-0003 is designated for use in modes 1-4, its logic is applicable to this EAL.

REFERENCES:

1. 0POP04-RC-0003, Rev. 18, Excessive RCS Leakage
2. 0POP02-RH-0001, Rev. 63, Residual Heat Removal System Operation

ECL: ALERT

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

Operating Mode Applicability: 5, 6, Defueled

Emergency Action Level:

Note: The Emergency Director should declare the ALERT 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 **ALL** three 4160V AC ESF Busses for 15 minutes or longer.

<u>Basis:</u>

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.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as a SITE AREA EMERGENCY because of the additional time available to restore an emergency bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition represents an actual or potential substantial degradation of the level of safety of the plant.

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

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC CS1 or RS1.

<u>CA2 – EAL-1 Selection Basis:</u>

N/A

<u>REFERENCES:</u>

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

CA3

ECL: ALERT

Initiating Condition: Inability to maintain the plant in cold shutdown.

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2)

Note: The Emergency Director should declare the ALERT promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

(1) UNPLANNED rise in RCS temperature to greater than 200 °F (Tavg) for greater than the duration specified in Table C3.

Table C3: RCS Heat-up Duration Thresholds

RCS Status	Containment Closure Status	Heat-up Duration
Intact (but not at reduced inventory)	Not applicable	60 minutes*
Not intact (or at reduced inventory)	Established	20 minutes*
	Not Established	0 minutes

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

(2) UNPLANNED RCS pressure rise greater than 10 psig. (This EAL does not apply during water-solid plant conditions.)

Basis: This IC addresses conditions involving a loss of decay heat removal capability or an addition of heat to the RCS in excess of that which can currently be removed. Either condition represents an actual or potential substantial degradation of the level of safety of the plant.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

EAL #1-The RCS Heat-up Duration Thresholds table addresses an rise in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not intact, or RCS inventory is reduced (e.g., mid-loop operation). The 20-minute criterion was included to allow time for operator action to address the temperature rise.

The RCS Heat-up Duration Thresholds table also addresses an rise in RCS temperature with the RCS intact. The status of CONTAINMENT CLOSURE is not crucial in this condition since the intact RCS is providing a high pressure barrier to a fission product release. The 60-minute time frame should allow sufficient time to address the temperature rise without a substantial degradation in plant safety.

Finally, in the case where there is a rise in RCS temperature, the RCS is not intact or is at reduced inventory and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because 1) the evaporated reactor coolant may be released directly into the Containment atmosphere and subsequently to the environment, and 2) there is reduced reactor coolant inventory above the top of irradiated fuel.

EAL #2- provides a pressure-based indication of RCS heat-up.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC CS1 or RS1.

CA3 – EAL-1 Selection Basis:

Table C3 was adopted from NEI 99-01, Rev. 6. This EAL addresses the concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal. A number of phenomena such as pressurization, vortexing, steam generator U-tube draining, RCS level differences when operating at a mid-loop condition, decay heat removal system design, and level instrumentation problems can lead to conditions where decay heat removal is lost and core uncover can occur. NRC analyses show that there are sequences that can cause core uncovery in 15 to 20 minutes, and severe core damage within an hour after decay heat removal is lost. The allowed time frames are consistent with the guidance provided by Generic Letter 88-17 and believed to be conservative given that a low pressure containment barrier to fission product release is established.

CA3 – EAL-2 Selection Basis:

An UNPLANNED RCS pressure rise greater than 10 psig provides a pressure-based indication of RCS heat-up. The pressure change, per NEI 99-01 Rev. 6, is the lowest change in pressure that can be accurately determined using installed instrumentation, but not less than 10 psig.

REFERENCES:

1. Technical Specifications Table 1.2 (Mode, Temperature, Power, keff Table)

CA6

ECL: ALERT

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

Operating Mode Applicability: 5, 6

Emergency Action Level:

(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

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 would be via IC CS1 or RS1.

CA6: EAL-1 Selection Basis:

The listed hazards are taken directly from NEI 99-01, Rev. 6. The only additional hazard was the inclusion of the Main Cooling Reservoir since it is a credible hazard and analyzed in the STPEGS UFSAR (reference 2).

REFERENCES:

1. STPEGS UFSAR, Rev. 13, Section 3.4.1, Flood Protection

ECL: SITE AREA EMERGENCY

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

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2 or 3)

Note: The Emergency Director should declare the SITE AREA EMERGENCY promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.

(1) a. CONTAINMENT CLOSURE not established.

AND

b. RCS level less than 33% of plenum.

(2) a. CONTAINMENT CLOSURE established.

AND

b. RCS level less than 0% of plenum

(3) a. RCS level cannot be monitored for 30 minutes or longer.

AND

b. Core uncovery is indicated by **ANY** of the following:

- Reactor Containment Building, 68'-0" Area Radiation Monitors RE-8055 or RE-8099 reading greater than 9,000 mR/hr.

  OR

- Erratic source range monitor indication.

  OR

- 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

Basis:

This IC addresses a significant and prolonged loss of reactor vessel/RCS inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a SITE AREA EMERGENCY declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in RCS level. If RCS level cannot be restored, fuel damage is probable.

Outage/shutdown contingency plans provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS levels of EALs 1.b and 2.b reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

In EAL 3.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

These EALs address concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.*

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC CG1 orRG1.

CS1: EAL-1 Selection Basis:

Per NEI 99-01 Rev. 6, the RCS level indication should be six inches (6") below the bottom inside diameter of the RCS loop penetration at the reactor vessel. Six inches (6") below the bottom inside diameter of the RCS hot leg nozzle (elevation 31'-0.5") is elevation 30'-6.5" per 0POP03-ZG-0009, Mid-Loop Operation, Addendum 1, RCS/RHR Simplified Elevation Diagram. The nearest RVWL Monitoring System thermocouples are located 6 inches above (Sensor 6) and 4.9 inches below (Sensor7) the prescribed elevation of 30'-6.5". When water level is at the desired elevation of 30'-6.5", Sensor 6 will be dry and Sensor 7 will be wet. This condition corresponds to a reading of 33% of plenum per 0POP02- II-0002, RVWL Monitoring System, Addendum 1, RVWL Sensor Elevations.

CS1: EAL-2 Selection Basis:

Per NEI 99-01 Rev. 6, the RCS level indication should be approximately the top of active fuel (TAF). The RCS level which corresponds to the top of the active fuel is 26'-1" . The nearest Reactor Vessel Water Level Monitoring System thermocouple to TAF is Sensor 8 at elevation 29'-2.7". Use of RVWL to approximate TAF; with the inherent gap of 37 inches between indicated level and actual level, is acceptable for the purposes of signaling that the threat to the public is reduced when CONTAINMENT CLOSURE is established.

CS1: EAL-3 Selection Basis:

As RCS level drops the dose rates above the core will rise. Area Radiation Monitors RE-8055 and RE-8099 are located on the 68'-0" elevation of the reactor containment building. Their locations are identified on drawing 9C129A81105. Their range (0.1 mR/hr to 10,000 mR/hr) is identified in Table 12.3.4-1 of Section 12 of the UFSAR. A rising trend on these monitors can be an indication that core uncovery is occurring. Additionally, erratic source range monitor indications, or large level rises in the tanks listed can give further indication of core uncovery.

The threshold value for radiation monitors RE-8055 and RE-8099 was based on Calculation STPNOC013-CALC-006 Rev.3. The calculated monitor response is 189 R/hr when RCS level is at the top of the active fuel. The high range of these monitors is 10 R/hr. The value of 9,000 mR/hr was selected to ensure that the threshold is readily assessable and within the calibrated range of the monitor. The threshold value of 9,000 mR/hr with the reactor head off corresponds to approximately 24 inches above the top of the active fuel; which provides an additional indication that RCS levels are near the point of fuel uncovery. These monitor readings in conjunction with the other threshold values allow for an accurate assessment of the EAL.

Core uncovery can be determined by the secondary indications listed in this EAL. The secondary indicators of inventory loss include a list of tanks/sumps found in 0POP04-RC-0003, Excessive RCS Leakage. Since other system leaks could raise levels in various tanks and sumps, the list has been limited to the tanks and sumps that would have the highest probability of indicating RCS leakage inside the Reactor Containment.

REFERENCES:

1. Calculation No: STPNOC013-CALC-006 Rev.3, Dose Rate Evaluation of Reactor Vessel Water Levels during Refueling for EAL Thresholds
2. 0POP03-ZG-0009, Rev. 59, Mid-Loop Operation, Addendum 1, RCS/RHR Simplified Elevation Diagram
3. USFAR, Rev. 15, Chapter 12, Table 12.3.4-1
4. 0POP02-II-0002, Rev. 15, RVWL Monitoring System
5. 0POP04-RC-0003, Rev 18, Excessive RCS Leakage
6. Drawing 9C129A81105, Re. 3, Radiation Zones, Reactor Containment Building, Plan at E. 68' – 0"

CG1

ECL: GENERAL EMERGENCY

Initiating Condition: Loss of RCS inventory affecting fuel clad integrity with containment challenged.

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2)

Note: The Emergency Director should declare the GENERAL EMERGENCY promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.

(1) a. RCS level less than 0% of plenum for 30 minutes or longer.

AND

b. **ANY** indication from the Table C1.

(2) a. RCS level cannot be monitored for 30 minutes or longer.

AND

b. Core uncovery is indicated by **ANY** of the following:

- Reactor Containment Building, 68'-0" Area Radiation Monitors RE-8055 or RE-8099 reading greater than 9,000 mR/hr.

  OR

- Erratic source range monitor indication

  OR

- UNPLANNED rise in **ANY** of the following sump or tank levels in Table C2 of sufficient magnitude to indicate core uncovery

AND

c. **ANY** indication from Table C1

Table C1: Containment Challenge
• CONTAINMENT CLOSURE not established * • ≥4% hydrogen exists inside containment • UNPLANNED rise in containment pressure

* IF CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, THEN declaration of a General Emergency is not required.

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

Basis:

This IC addresses the inability to restore and maintain RCS level above the top of active fuel with containment challenged. This condition represents actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in RCS level. If RCS level cannot be restored, fuel damage is probable.

With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a GENERAL EMERGENCY is not required.

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to Containment integrity.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use indications in Table C1 to assess whether or not containment is challenged.

In EAL 2.b, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS

These EALs address concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.*

CG1: EAL-1 Selection Basis:

Per NEI 99-01 Rev. 6, the RCS level indication should be approximately the top of active fuel (TAF). The RCS level which corresponds to the top of the active fuel is 26'-1". The nearest Reactor Vessel Water Level Monitoring System thermocouple to TAF is Sensor 8 at elevation 29'-2.7". Use of RVWL to approximate TAF; with the inherent gap of 37 inches between indicated level and actual level, is acceptable for the purposes of maintaining the escalation logic for the loss of RCS level condition.

CG1: EAL-2 Selection Basis:

The secondary indicators of inventory loss include a list of tanks/sumps found in 0POP04-RC-0003, Excessive RCS Leakage. Since other system leaks could rise levels in various tanks and sumps, the list has been limited to the tanks and sumps that would have the highest probability of indicating RCS leakage inside the Reactor Containment Building.

As RCS level drops the dose rates above the core will rise. Area Radiation Monitors RE-8055 and RE-8099 are located on the 68'-0" elevation of the reactor containment building. Their locations are identified on drawing 9C129A81105. Their range (0.1 mR/hr to 10,000 mR/hr) is identified in Table 12.3.4-1 of Section 12 of the UFSAR. Rising indication on these monitors forewarns core uncovery. Additionally, erratic source range monitor indications, or large level rises in the tanks listed can give further indication of core uncovery.

The threshold value for radiation monitors RE-8055 and RE-8099 was based on Calculation STPNOC013-CALC-006 Rev.3. The calculated monitor response is 189 R/hr when RCS level is at the top of the active fuel . The high range of these monitors is 10 R/hr. The value of 9,000 mR/hr was selected for this threshold to ensure the threshold is readily assessable and within the calibrated range of the monitor. The threshold value of 9,000 mR/hr with the reactor head off corresponds to approximately 24 inches above the top of the active fuel which provides an additional indication that RCS levels are near the point of fuel uncovery. These monitor readings in conjunction with the other threshold values allow for an accurate assessment of the EAL.

REFERENCES:

1. Calculation No. STPNOC013-CALC-006 Rev.3, Dose Rate Evaluation of Reactor Vessel Water Levels during Refueling for EAL Thresholds
2. 0POP03-ZG-0009, Rev. 59, Mid-Loop Operations
3. Drawing 9C129A81105, Rev. 3, Radiation Zones, Reactor Containment Building Plan at El. 68'-0"
4. USFAR, Rev. 15, Chapter 12, Table 12.3.4-1, Area Radiation Monitors
5. 0POP05-EO-E010, Rev. 21, Loss of Reactor or Secondary Coolant
6. 0POP04-RC-0003, Rev. 18, Excessive RCS Leakage

8 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) ICS/EALS

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

<u>UNUSUAL EVENT</u>

E-HU1 Damage to a loaded cask CONFINEMENT BOUNDARY.
*Op. Modes: **ALL***

E-HU1

ECL: UNUSUAL EVENT

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

Operating Mode Applicability: ALL

Emergency Action Level:

(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
b. 7000 mrem/hr *(gamma + neutron)* on the side surface of the transfer cask.

Basis:

This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage.

The existence of "damage" is determined by radiological survey. The values for this EAL are 2 times the Technical Specification allowable radiation levels. The technical specification multiple of "2 times", which is also used in Recognition Category R IC RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

Security-related events for ISFSIs are covered under ICs HU1 and HA1.

E-HU1 – EAL-1 Selection Basis:

NEI 99-01 Rev.6 states that the dose rate limits are 2 times the Cask Technical Specification Limits. Section 5.3.2 of the "Certificate of Compliance No. 1032, Appendix A, Technical Specifications For The HI-STORM FW MPC Storage System", states:

5.3.4 Notwithstanding the limits established in Section 5.3.3, the measured dose rates on a loaded OVERPACK or TRANSFER CASK shall not exceed the following values:

a. 30 mrem/hr (gamma + neutron) on the top of the OVERPACK

b. 300 mrem/hr (gamma + neutron) on the side of the OVERPACK, excluding inlet and outlet ducts

c. 3500 mrem/hr (gamma + neutron) on the side of the TRANSFER CASK

<u>REFERENCES:</u>

1. Certificate of Compliance no. 1032, Appendix A, Technical Specifications For The HI-STORM FW MPC Storage System, Section 5.3, Radiation Protection Program.10 CFR 72.104, Criteria For Radioactive Materials In Effluents And Direct Radiation From An ISFSI or MRS

9 FISSION PRODUCT BARRIER ICS/EALS

Table 9-F-1: Recognition Category "F" Initiating Condition Matrix

ALERT	
FA1	**ANY** Loss or **ANY** Potential Loss of either the Fuel Clad or RCS barrier. *Op. Modes: 1,2,3,4*
SITE AREA EMERGENCY	
FS1	Loss or Potential Loss of **ANY** two barriers. *Op. Modes: 1,2,3,4*
GENERAL EMERGENCY	
FG1	Loss of **ANY** two barriers and Loss or Potential Loss of the third barrier. *Op. Modes: 1,2,3,4*

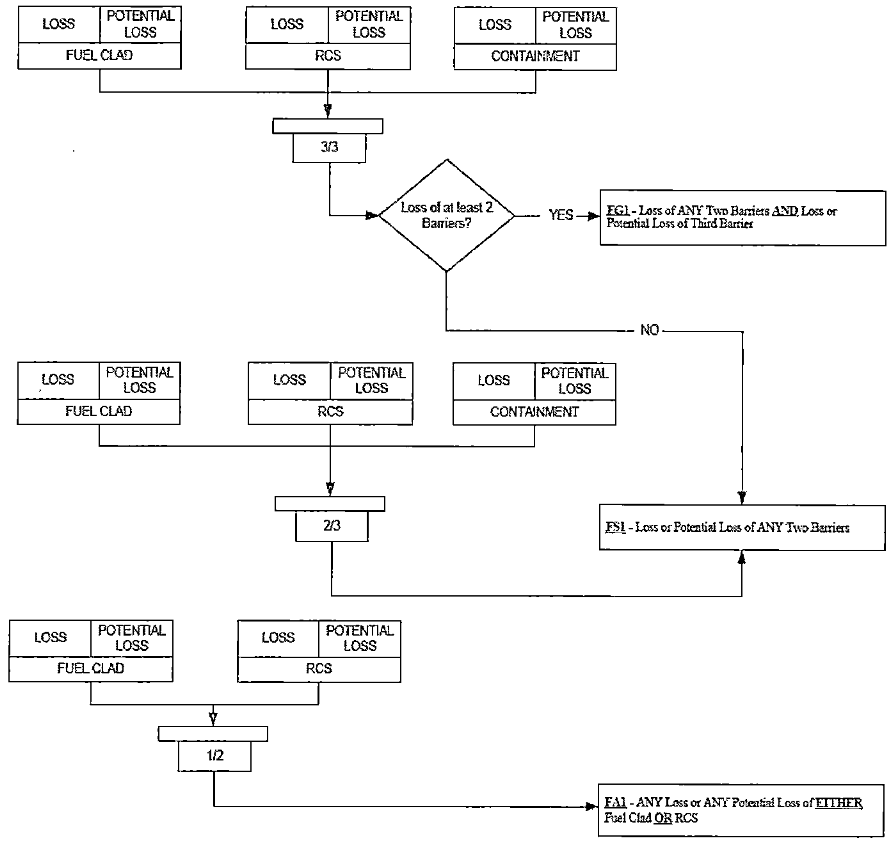


Table 9-F-2: EAL Fission Product Barrier Table

Thresholds for LOSS or POTENTIAL LOSS of Barriers

FA1 ALERT	**FS1 SITE AREA EMERGENCY**	**FG1 GENERAL EMERGENCY**
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.

Fuel Clad Barrier		**RCS Barrier**		**Containment Barrier**	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. RCS or SG Tube Leakage		**1. RCS or SG Tube Leakage**		**1. RCS or SG Tube Leakage**	
Not Applicable	A. Core Cooling - Orange entry conditions met	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	A. A leaking or RUPTURED SG is FAULTED outside of containment.	Not Applicable

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
2. Inadequate Heat Removal		**2. Inadequate Heat Removal**		**2. Inadequate Heat Removal**	
A. Core Cooling - Red entry conditions met	A. Core Cooling - Orange entry conditions met **OR** B. Heat Sink - Red entry conditions met	Not Applicable	A. Heat Sink - Red entry conditions met.	Not Applicable	A. Core Cooling – Red entry conditions met for 15 minutes or longer.
3. RCS Activity / Containment Radiation		**3. RCS Activity / Containment Radiation**		**3. RCS Activity / Containment Radiation**	
A1. RCB Rad Monitor RT-8050 or RT-8051 greater than 2100 R/hr **OR** 2. HATCH MONITOR greater than 4200mR/hr **OR** B. Sample analysis indicates that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131.	Not Applicable	A1. RCB Rad Monitor RT-8050 or RT-8051 greater than 10 R/hr **OR** 2. HATCH MONITOR greater than 20mR/hr	Not Applicable	Not Applicable	A1. RCB Rad Monitor RT-8050 or RT-8051 greater than 45,000 R/hr **OR** 2. HATCH MONITOR greater than 90,000mR/hr

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
4. Containment Integrity or Bypass		**4. Containment Integrity or Bypass**		**4. Containment Integrity or Bypass**	
Not Applicable	Not Applicable	Not Applicable	Not Applicable	A. Containment isolation is required **AND** EITHER of the following: 1. Containment integrity has been lost based on Emergency Director judgment. **OR** 2. UNISOLABLE pathway from the containment to the environment exists. **OR** B. Indications of RCS leakage outside of containment.	A. Containment - Red entry conditions met **OR** B. Explosive mixture exists inside containment ($H_2 \geq 4\%$) **OR** C1. Containment pressure greater than 9.5 psig. **AND** 2. Less than one full train of Containment Spray is operating per design for 15 minutes or longer.

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
5. Other Indications		**5. Other Indications**		**5. Other Indications**	
A. N/A	A. N/A	A. N/A	A. N/A	A. N/A	A. N/A
6. Emergency Director Judgment		**6. Emergency Director Judgment**		**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 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 Containment Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Basis Information For
EAL Fission Product Barrier Table 9-F-2

STP 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.

FUEL CLAD BARRIER THRESHOLDS

The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.

1. RCS or SG Tube Leakage

Loss 1

There is no Loss threshold associated with RCS or SG Tube Leakage.

Potential Loss 1.A

Core Cooling - Orange entry conditions (CETs ≥ 708° F) are sufficient to allow the onset of heat-induced cladding damage.

2. Inadequate Heat Removal

Loss 2.A

Core Cooling - Red entry conditions (CETs ≥ 1200° F) are sufficient to cause significant superheating of reactor coolant.

Potential Loss 2.A

Core Cooling - Orange entry conditions (CETs ≥ 708° F) are sufficient to allow the onset of heat-induced cladding damage.

Potential Loss 2.B

Heat Sink - Red entry conditions met (NR level in all SG ≤ 14% [34%] AND total AFW flow to SG ≤ 576 GPM). This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the Fuel Clad Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

Meeting this threshold results in a SITE AREA EMERGENCY because this threshold is identical to RCS Barrier Potential Loss threshold 2.A; both will be met. This condition warrants a SITE AREA EMERGENCY declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and raise RCS pressure to the point where mass will be lost from the system.

FUEL CLAD BARRIER THRESHOLDS

3. RCS Activity / Containment Radiation

Loss 3.A1

The readings for the containment high range area monitors (RT-8050 and RT-8051) correspond to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier. The values for RT-8050 and RT-8051 were based on Calculation15-RA-011. The threshold values were conservatively rounded down from the calculated value of 2144 R/hr to make the values readily assessable. Temperature induced current (TIC) limitations are not applicable to the Fuel Clad Barrier Loss threshold 3.A.1 because the expected radiation dose for this event overwhelms the TIC effect. This effect is discussed in the 10CFR50.59 evaluation 04-8245-60 associated with DCP 04-8245-33.

Loss 3.A2

The HATCH MONITOR is located outside containment and is the back-up monitor to the containment high range monitors (RT-8050 and RT-8051). The HATCH MONITOR threshold value is based on Calculation No. 03-ZE-003. This value corresponds to the calculated containment high range monitor readings for Fuel Clad Barrier Loss 3.A

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold 3.A since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the EMERGENCY CLASSIFICATION LEVEL to a SITE AREA EMERGENCY.

Loss 3.B

This threshold indicates that RCS radioactivity concentration is greater than 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

Potential Loss 3.

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

4. Containment Integrity or Bypass

Not Applicable (included for numbering consistency)

FUEL CLAD BARRIER THRESHOLDS

5. Other Indications

Loss and/or Potential Loss 5.A

N/A

6. Emergency Director Judgment

Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad Barrier is lost.

Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

RCS BARRIER THRESHOLDS

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

1. RCS or SG Tube Leakage

Loss 1.A

This threshold is based on an UNISOLABLE RCS leak of sufficient size to require an automatic or manual actuation of the Emergency Core Cooling System (ECCS). This condition clearly represents a loss of the RCS Barrier.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED. If a RUPTURED steam generator is also FAULTED outside of containment, the declaration escalates to a SITE AREA EMERGENCY since the Containment Barrier Loss threshold 1.A will also be met.

Potential Loss 1.A

This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ECCS (SI) actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs that a standby charging (makeup) pump be placed in service to restore and maintain pressurizer level.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

If a leaking steam generator is also FAULTED outside of containment, the declaration escalates to a SITE AREA EMERGENCY since the Containment Barrier Loss threshold 1.A will also be met.

Potential Loss 1.B

Integrity – Red entry conditions indicate an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized).

RCS BARRIER THRESHOLDS

2. Inadequate Heat Removal

Loss 2.A

There is no Loss threshold associated with Inadequate Heat Removal.

Potential Loss 2.A

Heat Sink – Red entry conditions met (NR level in all SG $\leq$ 14% [34%] AND total AFW flow to SGs $\leq$ 576 GPM).

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the RCS Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

Meeting this threshold results in a SITE AREA EMERGENCY because this threshold is identical to Fuel Clad Barrier Potential Loss threshold 2.B; both will be met. This condition warrants a SITE AREA EMERGENCY declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and raise RCS pressure to the point where mass will be lost from the system.

3. RCS Activity / Containment Radiation

Loss 3.A1.

Calculation 15-RA-11 provides a value that corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals the Technical Specification allowable limits. The threshold values were conservatively rounded down from the calculated value of 13 R/hr to make the values readily assessable. Temperature induced current (TIC) limitations are not applicable to the RCS Barrier Loss threshold 3.A1 because the expected radiation dose for this event overwhelms the TIC effect. This effect is discussed in the 10CFR50.59 evaluation 04-8245-60 associated with DCP 04-8245-33.

Loss 3.A2

The HATCH MONITOR is located outside containment and is the back-up monitor to the containment high range monitors (RT-8050 and RT-8051). The HATCH MONITOR threshold value is based on Calculation No. 03-ZE-003. This value corresponds to the calculated containment high range monitor readings for RCS Barrier Loss 3.A1

Potential Loss 3.

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

RCS BARRIER THRESHOLDS

4. Containment Integrity or Bypass

Not Applicable (included for numbering consistency)

5. Other Indications

Loss and/or Potential Loss 5.A

Variables used to monitor for the significant breach or the potential significant breach of fuel clad, the RCS pressure boundary, or the reactor Containment, are designated Type C. The response characteristics of Type C information display channels allow the control room operator to detect conditions indicative of significant failure of any of the three fission product barriers or the potential for significant failure of these barriers. Although variables selected to fulfill Type C functions may rapidly approach the values that indicate an actual significant failure, it is the final steady-state value reached that is important. Therefore, a high degree of accuracy and a rapid response time are not necessary for Type C information display channels. Type C variables are found in UFSAR Table 7B.6-1.

6. Emergency Director Judgment

Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is lost.

Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

CONTAINMENT BARRIER THRESHOLDS

The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the ECL from ALERT to a SITE AREA EMERGENCY or a GENERAL EMERGENCY.

1. RCS or SG Tube Leakage

<u>Loss 1.A</u>

This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for RCS Barrier Potential Loss 1.A and Loss 1.A, respectively. This condition represents a bypass of the containment barrier.

FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably [*part of the FAULTED definition*] and the FAULTED steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition, the steam generator is still considered FAULTED for emergency classification purposes.

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU3 for the fuel clad barrier (i.e., RCS activity values) and IC SU4 for the RCS barrier (i.e., RCS leak rate values).

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, glad seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

CONTAINMENT BARRIER THRESHOLDS

The EMERGENCY CLASSIFICATION LEVELS resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

	Affected SG is FAULTED Outside of Containment?	
P-to-S Leak Rate	**Yes**	**No**
Less than or equal to 25 gpm	No classification	No classification
Greater than 25 gpm	UNUSUAL EVENT per SU4	UNUSUAL EVENT per SU4
Requires operation of a standby charging pump (*RCS Barrier Potential Loss*)	SITE AREA EMERGENCY per FS1	ALERT per FA1
Requires an automatic or manual ECCS (SI) actuation (*RCS Barrier Loss*)	SITE AREA EMERGENCY per FS1	ALERT per FA1

Potential Loss 1.

There is no Potential Loss threshold associated with RCS or SG Tube Leakage.

2. Inadequate Heat Removal

Loss 2

There is no Loss threshold associated with Inadequate Heat Removal.

Potential Loss 2.A

Core Cooling – Red entry conditions met for 15 minutes or longer. This condition represents an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and a higher potential for containment failure. For this condition to occur there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.

The restoration procedure is considered "effective" if core exit thermocouple readings are decreasing and/or if RCS level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The Emergency Director should escalate the emergency classification level as soon as it is determined that the procedure(s) will not be effective.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation in a significant fraction of core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.

CONTAINMENT BARRIER THRESHOLDS

3. RCS Activity / Containment Radiation

Loss 3

There is no Loss threshold associated with RCS Activity / Containment Radiation.

Potential Loss 3.A.1

The readings for the containment high range area monitors (RT-8050 and RT-8051) correspond to an instantaneous release of the radioactive material inventory of the reactor coolant system (i.e., All the RCS coolant mass) into the containment, assuming that 20% of the fuel cladding has failed. The values for RT-8050 and RT-8051 were based on Calculation No.15-RA-11. The threshold values used were conservatively rounded down from the calculated value of 45,040 R/hr to ensure the values were readily assessable. This level of assumed fuel clad failure is well beyond that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds. Temperature induced current (TIC) limitations are not applicable to the Containment Barrier Potential Loss threshold 3.A.1 because the expected radiation dose for this event overwhelms the TIC effect. This effect is discussed in 10CFR50.59 evaluation 04-8245-60 associated with DCP 04-8245-33.

NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the EMERGENCY CLASSIFICATION LEVEL to a GENERAL EMERGENCY.

Potential Loss 3.A.2

The HATCH MONITOR is located outside containment and is the back-up monitor to the containment high range monitors (RT-8050 and RT-8051). The HATCH MONITOR threshold value is based on Calculation No. 03-ZE-003. This value corresponds to the calculated containment high range monitor readings for Containment Barrier Threshold Potential Loss 3.A.1.

4. Containment Integrity or Bypass

Loss 4.A

These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both thresholds 4.A.1 and 4.A.2.

4.A.1 – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate

during accident conditions, it is expected that the Emergency Director will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure 9-F-3. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

Another example would be a loss or potential loss of the RCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

4.A.2 – Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment atmosphere to the environment. As used here, the term "environment" includes the atmosphere of a room or area, outside the containment, that may, in turn, communicate with the outside-the-plant atmosphere (e.g., through discharge of a ventilation system or atmospheric leakage). Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.

Refer to the top piping run of Figure 9-F-3 in Addendum 3, Containment Integrity or Bypass Examples. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Leakage between two interfacing liquid systems, by itself, does not meet this threshold.

Refer to the bottom piping run of Figure 9-F-3. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the Component Cooling Water system to the Auxiliary Building, then no threshold has been met. If the pump or system piping developed a leak that allowed steam/water to enter the Auxiliary Building, then threshold 4.B would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause threshold 4.A.1 to be met as well.

CONTAINMENT BARRIER THRESHOLDS

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. Minor releases may also occur if a containment isolation valve(s) fails to close but the containment atmosphere escapes to a closed system. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold 1.A.

Loss 4.B

Containment sump, temperature, pressure and/or radiation levels will rise if reactor coolant mass is leaking into the containment. If these parameters have not risen, then the reactor coolant mass may be leaking outside of containment (i.e., a containment bypass sequence). Rises in sump, temperature, pressure, flow and/or radiation level readings outside of the containment may indicate that the RCS mass is being lost outside of containment.

Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of RCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside of containment may not rise significantly; however, other unexpected changes in sump levels, area temperatures or pressures, flow rates, etc. should be sufficient to determine if RCS mass is being lost outside of the containment.

Refer to the middle piping run of Figure 9-F-3. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause threshold 4.A.1 to be met as well.

To ensure proper escalation of the emergency classification, the RCS leakage outside of containment must be related to the mass loss that is causing the RCS Loss and/or Potential Loss threshold 1.A to be met.

Potential Loss 4.A

Containment – Red entry conditions met (containment pressure $\geq$ 56.5 PSIG). If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the RCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a SITE AREA EMERGENCY and GENERAL EMERGENCY since there is now a potential to lose the third barrier.

Potential Loss 4.B

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit (4%)). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a potential loss of the Containment Barrier.

CONTAINMENT BARRIER THRESHOLDS

Potential Loss 4.C

This threshold describes a condition where containment pressure is greater than the setpoint (9.5 PSIG) at which Containment Spray is 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. This threshold represents a potential loss of containment in that Containment Spray is either lost or performing in a degraded manner.

5. Other Indications

Loss and/or Potential Loss 5.A
N/A

6. Emergency Director Judgment

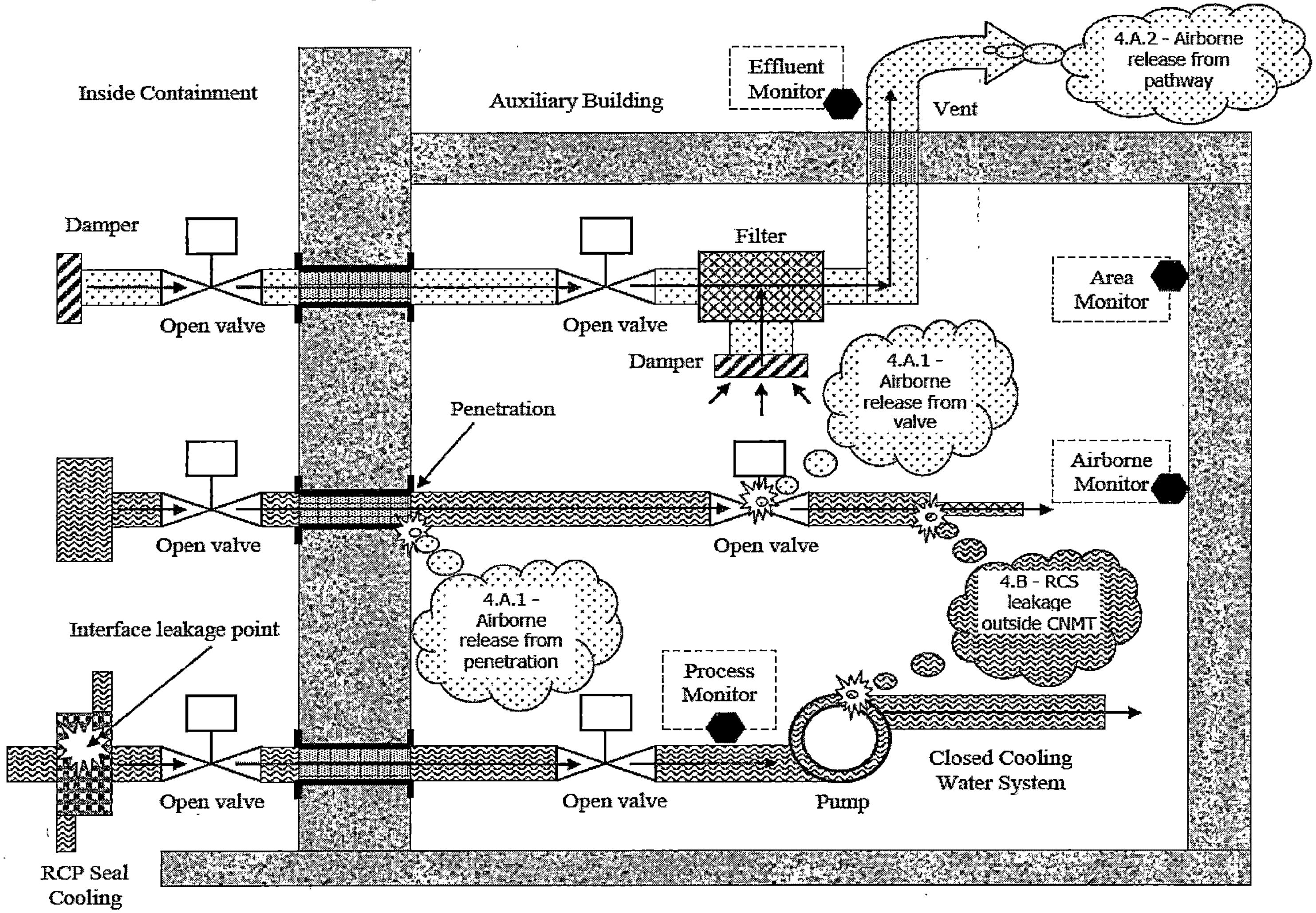
Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is lost.

Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Figure 9-F-3: Containment Integrity or Bypass Examples



NOTES: Only Supplemental Purge is a filtered release and STPEGS Component Cooling Water is equivalent to Closed Cooling Water

10 HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY ICS/EALS

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

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
HU1 Confirmed SECURITY CONDITION or threat. *Op. Modes: **ALL***	**HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes. *Op. Modes: **ALL***	**HS1** HOSTILE ACTION within the PROTECTED AREA. *Op. Modes: **ALL***	**HG1** HOSTILE ACTION resulting in loss of physical control of the facility. *Op. Modes: **ALL***
HU2 Seismic event greater than OBE levels. *Op. Modes: **ALL*** **HU3** Hazardous event. *Op. Modes: **ALL*** **HU4** FIRE potentially degrading the level of safety of the plant. *Op. Modes: **ALL***	**Note:** **See SA9 or CA6 for escalation of these events**		
	HA5 Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown. *Op. Modes: **ALL***		
	HA6 Control Room evacuation resulting in transfer of plant control to alternate locations. *Op. Modes: **ALL***	**HS6** Inability to control a key safety function from outside the Control Room. *Op. Modes: **ALL***	
HU7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT. *Op. Modes: **ALL***	**HA7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of an ALERT. *Op. Modes: **ALL***	**HS7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY. *Op. Modes: **ALL***	**HG7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY. *Op. Modes: **ALL***

ECL: UNUSUAL EVENT

Initiating Condition: Confirmed SECURITY CONDITION or threat.

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2 or 3)

(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

(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 SAFETY SYSTEM equipment, 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 ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Force 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 Security Force Supervisor 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 § 2.39039 information.

EAL #2- addresses the receipt of a CREDIBLE SECURITY THREAT. The credibility of the threat is assessed in accordance with 0SDP01-ZS-0011, Implementing Procedure For Safeguards Contingency Events.

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 0POP04-ZO-SEC4, Guideline For Airborne (Aircraft) Threat, and Security Force Instruction SI 2700, Security Response to Airborne Threat.

Emergency plans and implementing procedures are public documents; therefore, EALs 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 is contained in the Security Plan.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC HA1.

HU1: EAL-1 Selection Basis:

For EAL-1, the position of Security Force Supervisor was included since it is a 24-hour position. Normally the event would not be reported by the Acting Security Manager or Security Manager because the Acting Security Manager position is not normally activated until after an UNUSUAL EVENT has been declared, and the Security Manager position is not normally activated until after an ALERT has been declared. However, reporting by the Acting Security Manager or Security Manager was included in the event these positions are staffed under unusual circumstances.

REFERENCEs:

1. 0ERP01-ZV-SH03, Rev. 12, Acting Security Manager
2. 0ERP01-ZV-TS08, Rev. 16, Security Manager
3. 0POP04-ZO-SEC4, Rev. 10, Guideline For Airborne (Aircraft) Threat (SUNSI)
4. 0SDP01-ZS-0011, Implementing Procedure For Safeguards Contingency Events (Safeguards)
5. Security Force Instruction SI 2700, Security Response to Airborne Threat (SUNSI)

ECL: UNUSUAL EVENT

Initiating Condition: Seismic event greater than OBE levels.

Operating Mode Applicability: ALL

Emergency Action Level:

(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 manner deemed appropriate by the Shift Manager or Emergency Director.

Basis:

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE). An earthquake greater than an OBE but less than a Safe Shutdown Earthquake (SSE) should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., performs walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Although the "SEISMIC EVENT" alarm (0.02 g) in EAL 1.a is set below an O.B.E earthquake (0.05 g), it does provide an indication that a seismic event has occurred. In order to determine whether an O.B.E. earthquake occurred, additional indications may be needed. Determination per 0POP04-SY-001, Seismic Event is not practical if it takes longer than 15 minutes to perform.

Indications described in the EAL should be limited to those that are immediately available to Control Room personnel and which can be readily assessed. Indications available outside the Control Room and/or which require lengthy times to assess (e.g., processing of scratch plates or recorded data) should not be used. The goal is to specify indications that can be assessed within 15-minutes of the actual or suspected seismic event.

The EAL 1.b- statement is included to ensure that a declaration does not result from felt vibrations caused by a non-seismic source (e.g., a dropped heavy load). The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration. It is recognized that this alternate EAL wording may cause a site to declare an UNUSUAL EVENT while another site, similarly affected but with readily assessable OBE indications in the Control Room, may not.

Depending upon the plant mode at the time of the event, escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC CA6 or SA9.

HU2: EAL-1 Selection Basis:

STP does not have a readily available indication in the Control Room for determining if the site has experienced an OBE. The 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 STP has experienced an OBE.

STP has implemented the alternative EAL described in NEI 99-01 Developer Notes in conjunction with using the installed indication. EAL-1, b. allows the Shift Manager or Emergency Director to determine if a seismic event has taken place, taking into consideration the Seismic Event alarm, Control Room personnel feeling an actual or potential seismic event and other indications deemed appropriate.

REFERENCES:

1. 0POP04-SY-0001, Rev. 8, Seismic Event
2. NEI 99-01, Rev. 6, Development of Emergency Action Levels for Non-Passive Reactors.

HU3

ECL: UNUSUAL EVENT

Initiating Condition: Hazardous event.

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2 or 3 or 4 or 5)

Note: EAL #4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

(1) A tornado strike within the PROTECTED AREA.

(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.

(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).

(4) A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.

(5) Predicted or actual breach of Main Cooling Reservoir retaining dike along North Wall

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

EAL #1- addresses a tornado striking (touching down) within the PROTECTED AREA.

EAL #2- addresses flooding of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

EAL #3- addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.

EAL #4- addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site flooding caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road. This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around

the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.

EAL#5- the Main Cooling Reservoir breach along the north wall which was included because it is a credible hazard and analyzed in the STPEGS UFSAR.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be based on ICs in Recognition Categories R, F, S or C.

HU3: EAL-1, EAL-2, EAL-3, EAL-4 Selection Basis:

N/A

REFERENCE:

1. STPEGS UFSAR, Section 3.4.1, Flood Protection

ECL: UNUSUAL EVENT

Initiating Condition: FIRE potentially degrading the level of safety of the plant.

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2 or 3 or 4)

Note: The Emergency Director should declare the UNUSUAL EVENT promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

(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

b. The FIRE is located within **ANY** of the plant rooms or areas in Table H4:

Table H4: 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)

(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 H4:

AND

c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.

(3) A FIRE within the ISFSI **OR** plant PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.

(4) A FIRE within the ISFSI **OR** plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

EAL #1
The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

EAL #2
This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then EAL #1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

EAL #3
In addition to a FIRE addressed by EAL #1 or EAL #2, a FIRE within the plant or ISFSI PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety.

EAL #4
If a FIRE within the plant or ISFSI PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.

Basis-Related Requirements from Appendix R

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and EXPLOSIONS."
>
> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.
>
> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

Depending upon the plant mode at the time of the event, escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC CA6 or SA9.

HU4: EAL-1.b, EAL-2.b Selection Basis:

The plant areas or rooms listed contain SAFETY SYSTEM equipment.

REFERENCES:

1. 0PGP03-ZF-0001, Rev. 26, Fire Protection Program
2. STPEGS UFSAR, Rev. 16, Section 7.4, Systems Required for Safe Shutdown

HU7

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: ALL

Emergency Action Level:

(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 an UE.

HU7: EAL-1 Selection Basis:

N/A

REFERENCES:

N/A

HA1

ECL: ALERT

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

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2)

(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

(2) A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.

<u>Basis:</u>

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.

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 0POP04-ZO-SEC4, Guidelines for Airborne (Aircraft) Threat, and Security Force Instruction SI 2700, Security Response to Airborne Threat.

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 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 is contained in the Security Plan.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC HS1.

<u>HA1: EAL-1 and EAL-2 Selection Basis:</u>

The EALs are taken from NEI 99-01, Rev. 6. For EAL-1, the positions of Security Force Supervisor OR Acting Security Manager were included because either of these positions could be activated prior to meeting this EAL. The Security Force Supervisor is a 24-hour position and the normally the Acting Security Manager is activated after an UNUSUAL EVENT has been declared. The Security Manager is also included although this position is normally activated after an ALERT.

<u>REFERENCES:</u>

1. 0ERP01-ZV-SH03, Rev. 12, Acting Security Manager
2. 0ERP01-ZV-TS08, Rev. 16, Security Manager
3. 0POP04-ZO-SEC4, Rev. 10, Guideline For Airborne (Aircraft) Threat (SUNSI)
4. Security Force Instruction SI 2700, Security Response to Airborne Threat (SUNSI)

HA5

ECL: ALERT

Initiating Condition: Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown.

Operating Mode Applicability: ALL

Emergency Action Level:

Note: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

(1) a. Release of a toxic, corrosive, asphyxiant or flammable gas into the Control Room or **ANY** of the plant rooms or areas listed in Table H3/R2:

AND

b. Entry into the room or area is prohibited or impeded.

TABLE H3/R2: Plant Areas Requiring Access		
MODE 4	RCB	RHR Heat Exchanger Rooms
	MAB	51 ft Room 335
	EAB	Roof, MCC 1G8, 4.16KV Switchgear Rooms
MODE 5	EAB	4.16KV Switchgear Rooms

Basis:

This IC addresses an event involving a release of a hazardous gas that precludes or impedes access to equipment necessary to maintain normal plant operation, or required for a normal plant cooldown and shutdown. This condition represents an actual or potential substantial degradation of the level of safety of the plant.

An ALERT declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.

Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the Emergency Director's judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

An emergency declaration is not warranted if any of the following conditions apply.

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release).
- For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

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

This EAL does not apply to firefighting activities that automatically or manually activate a fire suppression system in an area.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via Recognition Category R, C or F ICs.

HA5: EAL-1 Selection Basis:

The areas listed in EAL-1 apply to areas that contain equipment necessary for plant operations, cooldown, or shutdown. Assuming all plant equipment is operating as designed, Normal operations and safe shutdown equipment operation is capable from the Main Control Room (MCR). The plant is able to transition into a hot shutdown from the MCR, therefore H3/R2 is a list of plant rooms or areas with entry-related mode applicability that contain equipment which require a manual/local action necessary following entry into hot shutdown (establish Residual Heat Removal shutdown cooling, disable operation of charging and ECCS equipment, and limit dilution pathways) and subsequent entry into cold shutdown (disable operation of ECCS equipment). After achieving cold shutdown it is assumed that the plant will be maintained in a cold shutdown condition.

REFERENCES:

1. 0PGP03-ZF-0001, Rev. 26, Fire Protection Program
2. STPEGS UFSAR, Rev. 16, Section 7.4, Systems Required for Safe Shutdown
3. 0POP03-ZG-0008, Rev. 56, Power Operations
4. 0POP03-ZG-0006, Rev. 54, Plant Shutdown from 100% to Hot Standby
5. 0POP03-ZG-0007, Rev. 71, Plant Cooldown

HA6

ECL: ALERT

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

Operating Mode Applicability: ALL

Emergency Action Level:

(1) An event has resulted in plant control being transferred from the Control Room to 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 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

HA7

ECL: ALERT

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

Operating Mode Applicability: ALL

Emergency Action Level:

(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.

HA7: EAL-1 Selection Basis:

N/A

REFERENCE:

N/A

ECL: SITE AREA EMERGENCY

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA.

Operating Mode Applicability: ALL

Emergency Action Level:

(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

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 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 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 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 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 is contained in the Security Plan.

Escalation of the 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

ECL: SITE AREA EMERGENCY

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

Operating Mode Applicability: ALL

Emergency Action Level:

Note: The Emergency Director should declare the SITE AREA EMERGENCY promptly upon determining that 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 the Auxiliary Shutdown Panel (ASP).

AND

b. Control of **ANY** of the following key safety functions in Table H2 is not reestablished within 15 minutes in Modes 1, 2 or 3 ONLY.

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 Auxiliary Shutdown Panel is based on Emergency Director judgment. The Emergency Director is expected to make a reasonable, informed judgment within 15 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 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. The mode applicability conditioning statement for Table H2 is based on the Technical Specification Operability requirement for the following functions of the Remote Shutdown System:

- Core reactivity control (initial and long term)

- RCS pressure control
- Decay heat removal via the AFW System and the SG safety valves or SG PORVs
- RCS inventory control via charging flow, and
- Safety support systems for the above functions.

REFERENCE:

1. Procedure 0POP04-ZO-0001, Rev. 35, Control Room Evacuation
2. Technical Specification 3.3.3.5 Remote Shutdown System

HS7

ECL: SITE AREA EMERGENCY

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

Operating Mode Applicability: ALL

Emergency Action Level:

(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.

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 SITE AREA EMERGENCY.

HS7: EAL-1 Selection Basis:

N/A

REFERENCE:

N/A

HG1

ECL: GENERAL EMERGENCY

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

Operating Mode Applicability: ALL

Emergency Action Level:

(1) a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by **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 in MODES 1, 2 or 3 ONLY.

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 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 Security Program]*.

Emergency plans and implementing procedures are public documents; therefore, EALs 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 is contained in 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. The mode applicability conditioning statement for Table H2 is based on the Technical Specification Operability requirement for the following Functions of the Remote Shutdown System:

- Core reactivity control (initial and long term)
- RCS pressure control
- Decay heat removal via the AFW System and the SG safety valves or SG PORVs
- RCS inventory control via charging flow, and
- Safety support systems for the above Functions.

REFERENCES:

1. 0ERP01-ZV-SH03, Rev. 12, Acting Security Manager
2. 0ERP01-ZV-TS08, Rev. 16, Security Manager
3. Technical Specification 3.3.3.5 Remote Shutdown System

ECL: GENERAL EMERGENCY

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

Operating Mode Applicability: ALL

Emergency Action Level:

(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.

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 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** offsite AC power capability to emergency buses for 15 minutes or longer. *Op. Modes: 1,2,3,4*	**SA1** Loss of **ALL** but one AC power source to emergency buses for 15 minutes or longer. *Op. Modes: 1,2,3,4*	**SS1** Loss of **ALL** offsite and **ALL** onsite AC power to emergency buses for 15 minutes or longer. *Op. Modes: 1,2,3,4*	**SG1** Prolonged loss of **ALL** offsite and **ALL** onsite AC power to emergency buses. *Op. Modes: 1,2,3,4*
SU2 UNPLANNED loss of Control Room indications for 15 minutes or longer. *Op. Modes: 1,2,3,4*	**SA2** UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress. *Op. Modes: 1,2,3,4*		
SU3 Reactor coolant activity greater than Technical Specification allowable limits. *Op. Modes: 1,2,3,4*			
SU4 RCS leakage for 15 minutes or longer. *Op. Modes: 1,2,3,4*			
SU5 Automatic or manual trip fails to shutdown the reactor. *Op. Modes: 1,2*	**SA5** 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. *Op. Modes: 1,2*	**SS5** Inability to shutdown the reactor causing a challenge to core cooling or RCS heat removal. *Op. Modes: 1,2*	

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

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

SU1

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

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

(1) Loss of **ALL** offsite AC power capability to **ALL** three 4160V AC ESF Buses 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 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

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

Note: The Emergency Director should declare the 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.

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 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 RCS level 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 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

SU3

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels: (1 or 2)

(1) RT-8039 reading greater than 30 $\mu Ci/cm^3$.

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

- Greater than 1μCi/gm Dose Equivalent I-131
- Greater than 100/ $\bar{E}$ 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 would be via ICs FA1 or the Recognition Category 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 $\mu Ci/cm^3$ 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/$\bar{E}$ bar μCi /gm gross activity.

REFERENCES:

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

SU4

ECL: UNUSUAL EVENT

Initiating Condition: RCS leakage for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels: (1 or 2 or 3)

Note: The Emergency Director should declare the 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 10 gpm for 15 minutes or longer.

(2) RCS identified leakage greater than 25 gpm for 15 minutes or longer.

(3) Leakage from the RCS to a location outside containment greater than 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) 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. An 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).

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

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via ICs of Recognition Category 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 Specification limit for primary-to-secondary leakage is 150 gallons per day through any one steam generator, but the specification does not specify the type of leakage. Therefore, STPEGS will use the leakage outside containment; which may include SG Tube Leakage, at 25 gpm for 15 minutes or longer in accordance with NEI 99-01 Rev. 6 guidance.

EFERENCES:

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

ECL: UNUSUAL EVENT

Initiating Condition: Automatic or manual trip fails to shutdown the reactor.

Operating Mode Applicability: 1, 2

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 did not shutdown the reactor.

AND

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

(2) a. A manual trip did not shutdown the reactor.

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

2. A subsequent automatic trip 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 that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control panels or an automatic trip 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, operators will promptly initiate manual actions at the reactor control panels to shutdown the reactor (e.g., initiate a manual reactor trip). 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 is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control panels to shut down the reactor (e.g., initiate a manual reactor trip) using a different switch). Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a

subsequent manual or automatic trip 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 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). 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 panels".

The plant response to the failure of an automatic or manual reactor trip 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 panels are also unsuccessful in shutting down the reactor, then the EMERGENCY CLASSIFICATION LEVEL will escalate to an 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 declaration is appropriate for this event.

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

Should a reactor trip 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 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 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%

SU6

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels: (1 or 2 or 3)

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

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

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

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

<u>**Basis:**</u>

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 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

ECL: UNUSUAL EVENT

Initiating Condition: Failure to isolate containment or loss of containment pressure control.

Operating Mode Applicability: 1, 2, 3, 4

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 within 15 minutes of the actuation signal.

(2) a. Containment pressure greater than 9.5 psig.

AND

b. No 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 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.

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) are either lost or performing in a degraded manner.

This event would escalate to a 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

ECL: ALERT

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

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

(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 a loss of **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 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 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 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

ECL: ALERT

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

Note: The Emergency Director should declare the 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.

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 trip
- ECCS (SI) actuation

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 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 RCS level 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 would be via ICs FS1 or IC 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

ECL: ALERT

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.

Operating Mode Applicability: 1, 2

Emergency Action Level:

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 did not shutdown the reactor.

AND

b. Manual actions taken at the reactor control 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 that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control 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 panels since this event entails a significant failure of the RPS.

A manual action at the reactor control 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). 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 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 panels".

The plant response to the failure of an automatic or manual reactor trip 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 or RCS heat removal safety functions, the EMERGENCY CLASSIFICATION LEVEL will escalate to a SITE AREA EMERGENCY via IC SS5. Depending upon plant responses and symptoms, escalation is also possible via IC FS1.

It is recognized that plant responses or symptoms may also require an 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

ECL: ALERT

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

(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

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 would be via IC FS1 or RS1.

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

ECL: SITE AREA EMERGENCY

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

Note: The Emergency Director should declare the 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 **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 would be via ICs RG1, FG1 or SG1.

SS1: EAL-1 Selection Criteria:

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

ECL: SITE AREA EMERGENCY

Initiating Condition: Inability to shutdown the reactor causing a challenge to core cooling or RCS heat removal.

Operating Mode Applicability: 1, 2

Emergency Action Level:

(1) a. An automatic or manual trip did not shutdown the reactor.

AND

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

AND

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

- Core Cooling – Red entry conditions met

  OR

- Heat Sink- Red entry conditions met

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip 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.

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 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 would be via IC RG1RG1 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.

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

ECL: SITE AREA EMERGENCY

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

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

(1) Indicated voltage is less than 105.5 VDC on **ALL** 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 would be via ICs 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

ECL: GENERAL EMERGENCY

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

Note: The Emergency Director should declare the GENERAL EMERGENCY promptly upon determining that 4 hours has been exceeded, or will likely be exceeded.

(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to **ALL** three 4160V AC ESF Buses.

AND

b. **EITHER** of the following:

- Restoration of at least one 4160VAC ESF bus in less than 4 hours is not likely.
- 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 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 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. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is a 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 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

ECL: GENERAL EMERGENCY

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

Note: The Emergency Director should declare the 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 **ALL** three 4160V AC ESF buses for 15 minutes or longer.

AND

b. Indicated voltage is less than 105.5 VDC on **ALL** 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

APPENDIX A – ACRONYMS AND ABBREVIATIONS

AC .... Alternating Current
AOP .... Abnormal Operating Procedure
ATWS .... Anticipated Transient Without Scram
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
ERG .... Emergency Response Guideline
FEMA .... Federal Emergency Management Agency
FSAR .... Final Safety Analysis Report
GE .... GENERAL EMERGENCY
IC .... Initiating Condition
ID .... Inside Diameter
ISFSI .... Independent Spent Fuel Storage Installation
Keff .... Effective Neutron Multiplication Factor
LCO .... Limiting Condition of Operation
LOCA .... Loss of Coolant Accident
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 .... Nuclear Management and Resources Council
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM .... Offsite Dose Calculation Manual
ORO .... Off-site Response Organization
PA .... Protected Area
PAG .... Protective Action Guideline
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment

PWR..Pressurized Water Reactor
PSIG..Pounds per Square Inch Gauge
R..Roentgen
RCS..Reactor Coolant System
Rem, rem, REM..Roentgen Equivalent Man
RPS..Reactor Protection System
RPV..Reactor Pressure Vessel
RVWL..Reactor Vessel Water Level
SAR..Safety Analysis Report
SCBA..Self-Contained Breathing Apparatus
SG..Steam Generator
SI..Safety Injection
SPDS..Safety Parameter Display System
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: 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: 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.

UNUSUAL EVENT UE: 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: 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 emergency classification scheme.

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 (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)

FISSION PRODUCT BARRIER THRESHOLD: A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.

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.

Selected terms used in INITIATING CONDITION and 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: The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.

CONTAINMENT CLOSURE: Those actions necessary to place the RCB in the closed containment condition that provides at least one integral barrier to the release of radioactive material. Sufficient separation of the containment atmosphere from the outside environment is to be provided such that a barrier to the escape of radioactive material is reasonably expected to remain in place following a core melt accident.

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

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: 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: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

REFUELING PATHWAY: 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.

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

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.

Attachment 3

STPEGS Emergency Action Level Technical Bases Document – redline markup

STPEGS Emergency Action Level Technical Bases Document Rev. 0

NEI 99-01 Rev. 6 Implementation

~~February~~ June 2015

NOTE: Changes to this document require a review under 10CFR50.54 (q) as directed by 0PGP05-ZV-0010, Emergency Plan Change.

TABLE OF CONTENTS

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1 DEVELOPMENT OF EMERGENCY ACTION LEVELS

1.1 REGULATORY BACKGROUND

Title 10, Code of Federal Regulations (CFR), Energy, contains the U.S. Nuclear Regulatory Commission (NRC) regulations that apply to nuclear power facilities. Several of these regulations govern various aspects of an emergency classification scheme. A review of the relevant sections listed below will aid the reader in understanding the key terminology provided in Section 3.0 of this document.

- 10 CFR § 50.47(a)(1)(i)
- 10 CFR § 50.47(b)(4)
- 10 CFR § 50.54(q)
- 10 CFR § 50.72(a)
- 10 CFR § 50, Appendix E, IV.B, Assessment Actions
- 10 CFR § 50, Appendix E, IV.C, Activation of Emergency Organization

The above regulations are supplemented by various regulatory guidance documents. Three documents of particular relevance to NEI 99-01 are:

- NUREG-0654/FEMA-REP-1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, October 1980. [Refer to Appendix 1, Emergency Action Level Guidelines for Nuclear Power Plants]

- NUREG-1022, Event Reporting Guidelines 10 CFR § 50.72 and § 50.73

Regulatory Guide 1.101, Emergency Response Planning and Preparedness for Nuclear Power Reactor

1.2 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

South Texas Project Electrical Generating Station (STP or STPEGS) is locating an ISFSI approximately 450 feet west of the Unit 2 Reactor Building. The STP ISFSI will be within the site Protected Area and is scheduled to be operational in 2016.

Selected guidance in NEI 99-01 is applicable to the STPEGS emergency plan to fulfill the requirements of 10 CFR 72.32 for a stand-alone ISFSI. The emergency classification levels applicable to an ISFSI are consistent with the requirements of 10 CFR § 50 and the guidance in NUREG 0654/FEMA-REP-1. The initiating conditions germane to a 10 CFR § 72.32 emergency plan (as described in NUREG-1567) are subsumed within the classification scheme for a 10 CFR § 50.47 emergency plan.

The STPEGS ICs and EALs for an ISFSI are presented in Section 8, ISFSI ICs/EALs. IC E-HU1 covers the spectrum of credible natural and man-made events included within the scope of the STPEGS ISFSI design. In addition, appropriate aspects of IC HU1 and IC HA1 address a HOSTILE ACTION directed against the STPEGS ISFSI.

The analysis of potential onsite and offsite consequences of accidental releases associated with the operation of an ISFSI is contained in NUREG-1140, *A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees*. NUREG-1140 concluded that the postulated worst-case accident involving an ISFSI has insignificant consequences to public health and safety. This evaluation shows that the maximum offsite dose to a member of the public due to an accidental release of radioactive materials would not exceed 1 rem Effective Dose Equivalent.

Regarding the above information, the expectations for an offsite response to an ALERT classified under a 10 CFR § 72.32 emergency plan are generally consistent with those for an UNUSUAL EVENT in a 10 CFR § 50.47 emergency plan (e.g., to provide assistance if requested). Also, the STPEGS Emergency Response Organization (ERO) required for a 10 CFR § 72.32 emergency plan is different than that prescribed for a 10 CFR § 50.47 emergency plan (e.g., no emergency technical support function).

1.3 NRC ORDER EA-12-051

The Fukushima Daiichi accident of March 11, 2012, was the result of a tsunami that exceeded the plant's design basis and flooded the site's emergency electrical power supplies and distribution systems. This caused an extended loss of power that severely compromised the key safety functions of core cooling and containment integrity, and ultimately led to core damage in three reactors. While the loss of power also impaired the spent fuel pool cooling function, sufficient water inventory was maintained in the pools to preclude fuel damage from the loss of cooling.

Following a review of the Fukushima Daiichi accident, the NRC concluded that several measures were necessary to ensure adequate protection of public health and safety under the provisions of the backfit rule, 10 CFR 50.109(a)(4)(ii). Among them was to provide each spent fuel pool with reliable level instrumentation to significantly enhance the ability of key decision-makers to allocate resources effectively following a beyond design basis event. To this end, the NRC issued Order EA-12-051, *Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation*, on March 12, 2012, to all US nuclear plants with an operating license, construction permit, or combined construction and operating license.

NRC Order EA-12-051 states, in part, "All licensees ... shall have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred." To this end, all licensees must provide:

- A primary and back-up level instrument that will monitor water level from the normal level to the top of the used fuel rack in the pool;
- A display in an area accessible following a severe event; and
- Independent electrical power to each instrument channel and provide an alternate remote power connection capability.

NEI 12-02, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation", provides guidance for complying with NRC Order EA-12-051.

This document includes three EALs that reflect the availability of the enhanced spent fuel pool level instrumentation associated with NRC Order EA-12-051. These EALs are included within existing IC RA2, and new ICs RS2 and RG2. These EALs will be implemented when the enhanced spent fuel pool level instrumentation is available for use.

2 KEY TERMINOLOGY USED

There are several key terms that appear throughout the EAL methodology. These terms are introduced in this section to support understanding of subsequent material. As an aid to the reader, the following table is provided as an overview to illustrate the relationship of the terms to each other.

<table>
<tr><td colspan="4">EMERGENCY CLASSIFICATION LEVEL</td></tr>
<tr><td>UNUSUAL EVENT</td><td>ALERT</td><td>SAE</td><td>GE</td></tr>
<tr><td>↓</td><td>↓</td><td>↓</td><td>↓</td></tr>
<tr><td>Initiating Condition</td><td>Initiating Condition</td><td>Initiating Condition</td><td>Initiating Condition</td></tr>
<tr><td>↓</td><td>↓</td><td>↓</td><td>↓</td></tr>
<tr><td>Emergency Action Level (1)
• Operating Mode Applicability
• Notes
• Basis</td><td>Emergency Action Level (1)
• Operating Mode Applicability
• Notes
• Basis</td><td>Emergency Action Level (1)
• Operating Mode Applicability
• Notes
• Basis</td><td>Emergency Action Level (1)
• Operating Mode Applicability
• Notes
• Basis</td></tr>
<tr><td colspan="4">(1) - When making an emergency classification, the Emergency Director must consider all information having a bearing on the proper assessment of an Initiating Condition. This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes and the informing Basis information. In the Recognition Category F matrices, EALs are referred to as Fission Product Barrier Thresholds; the thresholds serve the same function as an EAL.</td></tr>
</table>

2.1 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)

2.1.1 UNUSUAL EVENT (UE)

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.

Purpose: The purpose of this classification is to assure that the first step in future response has been carried out, to bring the operations staff to a state of readiness, and to provide systematic handling of unusual event information and decision-making.

2.1.2 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.

Purpose: The purpose of this classification is to assure that emergency personnel are readily available to respond if the situation becomes more serious or to perform confirmatory radiation monitoring if required, and provide offsite authorities current information on plant status and parameters.

2.1.3 SITE AREA EMERGENCY (SAE)

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.

Purpose: The purpose of the SITE AREA EMERGENCY declaration is to assure that emergency response centers are staffed, to assure that monitoring teams are dispatched, to assure that personnel required for evacuation of near-site areas are at duty stations if the situation becomes more serious, to provide consultation with offsite authorities, and to provide updates to the public through government authorities.

2.1.4 GENERAL EMERGENCY (GE)

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.

Purpose: The purpose of the GENERAL EMERGENCY declaration is to initiate predetermined protective actions for the public, to provide continuous assessment of information from the licensee and offsite organizational measurements, to initiate additional measures as indicated by actual or potential releases, to provide consultation with offsite authorities, and to provide updates for the public through government authorities.

2.2 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.

Discussion: An IC describes an event or condition, the severity or consequences of which meets the definition of an emergency classification level. An IC can be expressed as a continuous, measurable parameter (e.g., RCS leakage), an event (e.g., an earthquake) or the status of one or more fission product barriers (e.g., loss of the RCS barrier).

Appendix 1 of NUREG-0654 does not contain example Emergency Action Levels (EALs) for each ECL, but rather Initiating Conditions (i.e., plant conditions that indicate that a radiological emergency, or events that could lead to a radiological emergency, has occurred). NUREG-0654 states that the Initiating Conditions form the basis for establishment by a licensee of the specific plant instrumentation readings (as applicable) which, if exceeded, would initiate the emergency classification. Thus, it is the specific instrument readings that would be the EALs.

Considerations for the assignment of a particular INITIATING CONDITION to an EMERGENCY CLASSIFICATION LEVEL are discussed in Section 3.

2.2.1 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.

Discussion: EAL statements may utilize a variety of criteria including instrument readings and status indications; observable events; results of calculations and analyses; entry into particular procedures; and the occurrence of natural phenomena.

2.2.2 FISSION PRODUCT BARRIER THRESHOLD

A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.

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

- Fuel Clad
- Reactor Coolant System (RCS)
- Containment

Upon determination that one or more fission product barrier thresholds have been exceeded, the combination of barrier loss and/or potential loss thresholds is compared to the fission product barrier IC/EAL criteria to determine the appropriate ECL.

In some accident sequences, the ICs and EALs presented in the Abnormal Radiation Levels/ Radiological Effluent (R) Recognition Category will be exceeded at the same time, or shortly after, the loss of one or more fission product barriers. This redundancy is intentional as the former ICs address radioactivity releases that result in certain offsite doses from whatever cause, including events that might not be fully encompassed by fission product barriers (e.g., spent fuel pool accidents, design containment leakage following a LOCA, etc.).

3 DESIGN OF THE STPEGS EMERGENCY CLASSIFICATION SCHEME

3.1 ASSIGNMENT OF EMERGENCY CLASSIFICATION LEVELS (ECLS)

An effective emergency classification scheme must incorporate a realistic and accurate assessment of risk, both to plant workers and the public. There are obvious health and safety risks in underestimating the potential or actual threat from an event or condition; however, there are also risks in overestimating the threat as well (e.g., harm that may occur during an evacuation). The emergency classification scheme attempts to strike an appropriate balance between reasonably anticipated event or condition consequences, potential accident trajectories, and risk avoidance or minimization.

There are a range of "non-emergency events" reported to the US Nuclear Regulatory Commission (NRC) staff in accordance with the requirements of 10 CFR § 50.72. Guidance concerning these reporting requirements, and example events, are provided in NUREG-1022. Certain events reportable under the provisions of 10 CFR § 50.72 may also require the declaration of an emergency.

In order to align each Initiating Conditions (IC) with the appropriate ECL, it was necessary to determine the attributes of each ECL. The goal of this process is to answer the question, "What events or conditions should be placed under each ECL?" The following sources provided information and context for the development of ECL attributes.

- Assessments of the effects and consequences of different types of events and conditions
- STPEGS abnormal and emergency operating procedure setpoints and transition criteria
- STPEGS Technical Specification limits and controls
- STPEGS Offsite Dose Calculation Manual (ODCM) radiological release limits
- Review of selected STPEGS Updated Final Safety Analysis Report (UFSAR) accident analyses
- Environmental Protection Agency (EPA) Protective Action Guidelines (PAGs)
- NUREG 0654, Appendix 1, Emergency Action Level Guidelines for Nuclear Power Plants
- Industry Operating Experience
- Input from subject matter experts at STPEGS

The following ECL attributes were created to aid in the development of ICs and Emergency Action Levels (EALs). The attributes may be useful in briefing and training settings (e.g., helping an Emergency Director understand why a particular condition is classified as an ALERT).

The attributes of each ECL are presented below.

3.1.1 UNUSUAL EVENT (UE)

An UNUSUAL EVENT, as defined in section 2.1.1, includes but is not limited to an event or condition that involves:

(A) A precursor to a more significant event or condition.
(B) A minor loss of control of radioactive materials or the ability to control radiation levels within the plant.
(C) A consequence otherwise significant enough to warrant notification to local, State and Federal authorities.

3.1.2 ALERT

An ALERT, as defined in section 2.1.2, includes but is not limited to an event or condition that involves:

(A) A loss or potential loss of either the fuel clad or Reactor Coolant System (RCS) fission product barrier.
(B) An event or condition that significantly reduces the margin to a loss or potential loss of the fuel clad or RCS fission product barrier.
(C) A significant loss of control of radioactive materials resulting in an inability to control radiation levels within the plant, or a release of radioactive materials to the environment that could result in doses greater than 1% of an EPA PAG at or beyond the site boundary.
(D) A HOSTILE ACTION occurring within the OWNER CONTROLLED AREA.

3.1.3 SITE AREA EMERGENCY (SAE)

A SITE AREA EMERGENCY, as defined in section 2.1.3, includes but is not limited to an event or condition that involves:

(A) A loss or potential loss of any two fission product barriers - fuel clad, RCS and/or containment.
(B) A precursor event or condition that may lead to the loss or potential loss of multiple fission product barriers within a relatively short period of time. Precursor events and conditions of this type include those that challenge the monitoring and/or control of multiple safety systems.
(C) A release of radioactive materials to the environment that could result in doses greater than 10% of an EPA PAG at or beyond the site boundary.
(D) A HOSTILE ACTION occurring within the plant PROTECTED AREA.

3.1.4 GENERAL EMERGENCY (GE)

A GENERAL EMERGENCY, as defined in section 2.1.4, includes but is not limited to an event or condition that involves:

(A) Loss of any two fission product barriers AND loss or potential loss of the third barrier - fuel clad, RCS and/or containment.
(B) A precursor event or condition that, unmitigated, may lead to a loss of all three fission product barriers. Precursor events and conditions of this type include those that lead directly to core damage and loss of containment integrity.
(C) A release of radioactive materials to the environment that could result in doses greater than an EPA PAG at or beyond the site boundary.
(D) A HOSTILE ACTION resulting in the loss of key safety functions (reactivity control, core cooling/RPV water level or RCS heat removal) or damage to spent fuel.

3.1.5 Risk-Informed Insights

Emergency preparedness is a defense-in-depth measure that is independent of the assessed risk from any particular accident sequence; however, the development of an effective emergency classification scheme can benefit from a review of risk-based assessment results. To that end, the development and assignment of certain ICs and EALs also considered insights from several site-specific probabilistic safety assessments (PSA - also known as probabilistic risk assessment, PRA). Some generic insights from this review included:

1. Accident sequences involving a prolonged loss of all AC power are significant contributors to core damage frequency. For this reason, a loss of all AC power for greater than 15 minutes, with the plant at or above Hot Shutdown, was assigned an ECL of SITE AREA EMERGENCY. Precursor events to a loss of all AC power were also included as an UNUSUAL EVENT and an ALERT.

   A station blackout coping analyses performed in response to 10 CFR § 50.63 and Regulatory Guide 1.155, *Station Blackout*, may be used to determine a time-based criterion to demarcate between a SITE AREA EMERGENCY and a GENERAL EMERGENCY. The time dimension is critical to a properly anticipatory emergency declaration since the goal is to maximize the time available for State and local officials to develop and implement offsite protective actions. 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.

2. For severe core damage events, uncertainties exist in phenomena important to accident progressions leading to containment failure. Because of these uncertainties, predicting the status of containment integrity may be difficult under severe accident conditions. This is why maintaining containment integrity alone following sequences leading to severe core damage is an insufficient basis for not escalating to a GENERAL EMERGENCY.

3. PSAs indicated that leading contributors to latent fatalities were sequences involving a containment bypass, a large Loss of Coolant Accident (LOCA) with early containment failure, a Station Blackout lasting longer than four hours, and a reactor coolant pump seal failure. The generic EAL methodology needs to be sufficiently rigorous to address these sequences in a timely fashion.

3.2 TYPES OF INITIATING CONDITIONS AND EMERGENCY ACTION LEVELS

The STPEGS methodology makes use of symptom-based, barrier-based and event-based ICs and EALs. Each type is discussed below.

Symptom-based ICs and EALs are parameters or conditions that are measurable over some range using plant instrumentation (e.g., core temperature, reactor coolant level, radiological effluent, etc.). When one or more of these parameters or conditions are off-normal, reactor operators will implement procedures to identify the probable cause(s) and take corrective action.

Fission product barrier-based ICs and EALs are the subset of symptom-based EALs that refer specifically to the level of challenge to the principal barriers against the release of radioactive material from the reactor core to the environment. These barriers are the fuel cladding, the reactor coolant system pressure boundary, and the containment. The barrier-based ICs and EALs consider the level of challenge to each individual barrier - potentially lost and lost - and the total number of barriers under challenge.

Event-based ICs and EALs define a variety of specific occurrences that have potential or actual safety significance. These include the failure of an automatic reactor scram/trip to shut down the reactor, natural phenomena (e.g., an earthquake), or man-made hazards such as a toxic gas release.

3.3 STPEGS DESIGN CONSIDERATIONS

The South Texas Project Electrical Generating Station (STPEGS) is composed of two units, each having an identical pressurized water reactor (PWR) Nuclear Steam Supply System (NSSS) and turbine generator (TG).

The NSSS is a Westinghouse Electric Corporation four-loop PWR. High-pressure light water serves as the coolant, neutron moderator, reflector, and solvent for the neutron absorber. The Reactor Coolant System (RCS), comprised of four parallel loops (each with a RCP and a steam generator [SG]), is used to transfer the heat generated in the core to the SGs using RCPs to circulate the water. RCS pressure is maintained by means of a pressurizer attached to the hot leg of one of the loops. The RCS is designed to circulate borated demineralized water at temperatures, pressures and flow rates consistent with the design thermal and hydraulic performance of the NSSS.

The Reactor Coolant Pressure Boundary Leak Detection System consists of temperature, level, humidity, and radioactivity sensors with associated instrumentation and alarms. Small leaks are detected by temperature and level changes of systems, increasing sump levels, and humidity and radioactivity concentration changes inside the Containment. Large leaks are detected by changes in reactor coolant inventory, changes in flow rates in process lines and changes in sump level.

Emergency Core Cooling System consists of three independent trains, each one capable of providing 100 percent of the required flow to the core in the unlikely event of a LOCA. Each train consists of one high-head safety injection pump and one low-head safety injection pump. Heat is removed from the system during recirculation by the residual heat removal heat exchanger (low-head pump only). The piping and valving associated with each of the three subsystems are identical. In the event of a steam pipe rupture, the ECCS provides adequate shutdown capability.

The Reactor Containment is a post-tensioned concrete cylinder with a steel liner plate, hemispherical top, and flat bottom. This structure provides a virtually leaktight barrier to prevent escape of fission products to the environment in the unlikely event of a loss of coolant accident (LOCA).

3.4 ORGANIZATION AND PRESENTATION OF GENERIC INFORMATION

The scheme's generic information is organized by Recognition Category in the following order.

- R- Abnormal Radiation Levels / Radiological Effluent – Section 6
- C - Cold Shutdown / Refueling System Malfunction – Section 7
- E - Independent Spent Fuel Storage Installation (ISFSI) – Section 8
- F - Fission Product Barrier – Section 9
- H - Hazards and Other Conditions Affecting Plant Safety – Section 10
- S - System Malfunction – Section 11

Each Recognition Category section contains a matrix showing the ICs and their associated EMERGENCY CLASSIFICATION LEVELS. The following information and guidance is provided for each IC:

- ECL – the assigned emergency classification level for the IC.
- Initiating Condition – provides a summary description of the emergency event or condition.
- Operating Mode Applicability – Lists the modes during which the IC and associated EAL(s) are applicable (i.e., are to be used to classify events or conditions).
- Emergency Action Level(s) – Provides indications that are considered to meet the intent of the IC.

For Recognition Category F, the fission product barrier thresholds are presented in tables and arranged by fission product barrier and the degree of barrier challenge (i.e., potential loss or loss). This presentation method shows the synergism among the thresholds, and supports accurate assessments.

Basis – Provides background information that explains the intent and application of the IC and EALs. In some cases, the basis also includes relevant source information and references.

3.5 IC AND EAL MODE APPLICABILITY

The STPEGS emergency classification scheme was developed recognizing that the applicability of ICs and EALs will vary with plant mode. For example, some symptom-based ICs and EALs can be assessed only during the power operations, startup, or hot standby/shutdown modes of operation when all fission product barriers are in place, and plant instrumentation and safety systems are fully operational. In the cold shutdown and refueling modes, different symptom-based ICs and EALs will come into play to reflect the opening of systems for routine maintenance, the unavailability of some safety system components and the use of alternate instrumentation.

The following table shows which Recognition Categories are applicable in each plant mode. The ICs and EALs for a given Recognition Category are applicable in the indicated modes.

MODE OF APPLICABILITY MATRIX

	Recognition Category					
Mode	**R**	**C**	**E**	**F**	**H**	**S**
Power Operations	X		X	X	X	X
Startup	X		X	X	X	X
Hot Standby	X		X	X	X	X
Hot Shutdown	X		X	X	X	X
Cold Shutdown	X	X	X		X	
Refueling	X	X	X		X	
Defueled	X	X	X		X	

STPEGS Operating Modes

Mode	**Description**	**Criteria (Rx Power excludes decay heat)**
1	Power Operations	Reactor Power > 5%, Keff ≥ 0.99 T Avg ≥ 350°F
2	Startup	Reactor Power ≤ 5%, Keff ≥ 0.99 T Avg ≥ 350°F
3	Hot Standby	Reactor Power 0% Keff < 0.99 T Avg ≥ 350°F
4	Hot Shutdown	Reactor Power 0% Keff < 0.99 350°F > T Avg > 200°F
5	Cold Shutdown	Reactor Power 0% Keff < 0.99 T Avg ≤ 200°F
6	Refueling	Reactor Power 0% Keff ≤ 0.95 T Avg ≤ 140°F Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.
	Defueled	All fuel removed from the reactor vessel (i.e., full core offload during refuel or extended outage)

4 STPEGS SCHEME DEVELOPMENT

4.1 GENERAL DEVELOPMENT PROCESS

The STPEGS ICs and EALs were developed to be unambiguous and readily assessable because both serve specific purposes. The IC is the fundamental event or condition requiring a declaration. The EAL(s) is the pre-determined threshold that defines when the IC is met. To this end, the STPEGS ICs and EALs were developed with input from key stakeholders such as Operations, Training, Health Physics, and Engineering. STPEGS specific indications, parameters and values were consistent with licensing basis documents, plant procedures, training, calculations, and drawings

Useful acronyms and abbreviations associated with the STPEGS emergency classification scheme are presented in Appendix A, Acronyms and Abbreviations. Those specific to STPEGS were included to be consistent with site terminology, site procedure, and training.

Many words or terms used in the STPEGS emergency classification scheme have scheme-specific definitions. These words and terms are identified by being set in all capital letters (i.e., ALL CAPS). The definitions are presented in Appendix B, Definitions.

4.2 CRITICAL CHARACTERISTICS

When crafting the scheme, STPEGS ensured that certain critical characteristics were met. These critical characteristics are listed below.

- The ICs, EALs, Operating Mode Applicability criteria, Notes and Basis information are consistent with industry guidance; while the actual wording may be different from NEI 99-01 Revision 6, the classification intent is maintained. With respect to Recognition Category F, the STPEGS scheme included a user-aid to facilitate timely and accurate classification of fission product barrier losses and/or potential losses. The user-aid logic is consistent with the classification logic presented in Section 9.
- EAL statements use objective criteria and observable values.
- ICs, EALs, Operating Mode Applicability and Note statements and formatting consider human factors and are user-friendly.
- The scheme facilitates upgrading and downgrading of the emergency classification where necessary.
- The scheme facilitates classification of multiple concurrent events or conditions.

4.3 INSTRUMENTATION USED FOR EALS

STPEGS incorporated instrumentation that is reliable and routinely maintained in accordance with site programs and procedures. Alarms referenced in EAL statements are those that are the most operationally significant for the described event or condition. EAL setpoints are within the calibrated range of the referenced instrumentation, and consider any automatic instrumentation functions that may impact accurate EAL assessment. In addition, EAL setpoint values do not use terms such as “off-scale low” or “off-scale high” since that type of reading may not be readily differentiated from an instrument failure. If instrumentation failures occur that have EALs associated with them (i.e., process radiation monitors) compensatory means of implementation may be used as described in plant procedures.

4.4 REFERENCES TO STPEGS AOPS AND EOPS

Some of the criteria/values used in several EALs and fission product barrier thresholds were drawn from STPEGS AOPs and EOPs. This approach was intended to maintain good alignment between operational diagnoses and emergency classification assessments. STPEGS verified the appropriate administrative controls are in place to ensure that a subsequent change to an AOP or EOP is screened to determine if an evaluation pursuant to 10 CFR 50.54(q) is required.

5 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

5.1 GENERAL CONSIDERATIONS

When making an emergency classification, the Emergency Director must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes and the informing Basis information. In the Recognition Category F matrices, EALs are referred to as Fission Product Barrier Thresholds; the thresholds serve the same function as an EAL.

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, Interim Staff Guidance, *Emergency Planning for Nuclear Power Plants*.

All emergency classification assessments should be based upon VALID indications, reports or conditions. A VALID indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy. For example, validation could be accomplished through an instrument channel check, response on related or redundant indicators, or direct observation by plant personnel. The validation of indications should be completed in a manner that supports timely emergency declaration.

For ICs and EALs that have a stipulated time duration (e.g., 15 minutes, 30 minutes, etc.), the Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time. If an ongoing radiological release is detected and the release start time is unknown, it should be assumed that the release duration specified in the IC/EAL has been exceeded, absent data to the contrary.

A planned work activity that results in an expected event or condition which meets or exceeds an EAL does not warrant an emergency declaration provided that 1) the activity proceeds as planned and 2) the plant remains within the limits imposed by the operating license. Such activities include planned work to test, manipulate, repair, maintain or modify a system or component. In these cases, the controls associated with the planning, preparation and execution of the work will ensure that compliance is maintained with all aspects of the operating

license provided that the activity proceeds and concludes as expected. Events or conditions of this type may be subject to the reporting requirements of 10 § CFR 50.72.

The assessment of some EALs is based on the results of analyses that are necessary to ascertain whether a specific EAL threshold has been exceeded (e.g., dose assessments, chemistry sampling, RCS leak rate calculation, etc.); the EAL and/or the associated basis discussion will identify the necessary analysis. In these cases, the 15-minute declaration period starts with the availability of the analysis results that show the threshold to be exceeded (i.e., this is the time that the EAL information is first available). The NRC expects licensees to establish the capability to initiate and complete EAL-related analyses within a reasonable period of time (e.g., maintain the necessary expertise on-shift).

While the EALs have been developed to address a full spectrum of possible events and conditions which may warrant emergency classification, a provision for classification based on operator/management experience and judgment is still necessary. This scheme provides the Emergency Director with the ability to classify events and conditions based upon judgment using EALs that are consistent with the Emergency Classification Level (ECL) definitions (refer to Category H). The Emergency Director will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated into the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

5.2 CLASSIFICATION METHODOLOGY

To make an emergency classification, the user will compare an event or condition (i.e., the relevant plant indications and reports) to an EAL(s) and determine if the EAL has been met or exceeded. The evaluation of an EAL(s) must be consistent with the related Operating Mode Applicability and Notes. If an EAL has been met or exceeded, then the IC is considered met and the associated ECL is declared in accordance with plant procedures.

When assessing an EAL that specifies a time duration for the off-normal condition, the "clock" for the EAL time duration runs concurrently with the emergency classification process "clock." For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01.

5.3 CLASSIFICATION OF MULTIPLE EVENTS AND CONDITIONS

When multiple emergency events or conditions are present, the user will identify all met or exceeded EALs. The highest applicable ECL identified during this review is declared. For example:

- If an ALERT EAL and a SITE AREA EMERGENCY EAL are met, whether at one unit or at two different units, a SITE AREA EMERGENCY should be declared.

There is no "additive" effect from multiple EALs meeting the same ECL. For example:

- If two ALERT EALs are met, whether at one unit or at two different units, an ALERT should be declared.

Related guidance concerning classification of rapidly escalating events or conditions is provided in Regulatory Issue Summary (RIS) 2007-02, *Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events*.

5.4 CONSIDERATION OF MODE CHANGES DURING CLASSIFICATION

The mode in effect at the time that an event or condition occurred, and prior to any plant or operator response, is the mode that determines whether or not an IC is applicable. If an event or condition occurs, and results in a mode change before the emergency is declared, the emergency classification level is still based on the mode that existed at the time that the event or condition was initiated (and not when it was declared). Once a different mode is reached, any new event or condition, not related to the original event or condition, requiring emergency classification should be evaluated against the ICs and EALs applicable to the operating mode at the time of the new event or condition.

For events that occur in Cold Shutdown or Refueling, escalation is via EALs that are applicable in the Cold Shutdown or Refueling modes, even if Hot Shutdown (or a higher mode) is entered during the subsequent plant response. In particular, the fission product barrier EALs are applicable only to events that initiate in the Hot Shutdown mode or higher.

5.5 CLASSIFICATION OF IMMINENT CONDITIONS

Although EALs provide specific thresholds, the Emergency Director must remain alert to events or conditions that could lead to meeting or exceeding an EAL within a relatively short period of time (i.e., a change in the ECL is IMMINENT). If, in the judgment of the Emergency Director, meeting an EAL is IMMINENT, the emergency classification should be made as if the EAL has been met. While applicable to all EMERGENCY CLASSIFICATION LEVELS, this approach is particularly important at the higher EMERGENCY CLASSIFICATION LEVELS since it provides additional time for implementation of protective measures.

5.6 EMERGENCY CLASSIFICATION LEVEL UPGRADING AND DOWNGRADING

An ECL may be downgraded when the event or condition that meets the highest IC and EAL no longer exists, and other site-specific downgrading requirements are met. If downgrading the ECL is deemed appropriate, the new ECL would then be based on a lower applicable IC(s) and EAL(s). The ECL may also simply be terminated.

The following approach to downgrading or terminating an ECL is recommended.

ECL	Action When Condition No Longer Exists
UNUSUAL EVENT	Terminate the emergency in accordance with plant procedures.
ALERT	Downgrade or terminate the emergency in accordance with plant procedures.
SITE AREA EMERGENCY with no long-term plant damage	Downgrade or terminate the emergency in accordance with plant procedures.
SITE AREA EMERGENCY with long-term plant damage	Terminate the emergency and enter recovery in accordance with plant procedures.
GENERAL EMERGENCY	Terminate the emergency and enter recovery in accordance with plant procedures.

As noted above, guidance concerning classification of rapidly escalating events or conditions is provided in RIS 2007-02.

5.7 CLASSIFICATION OF SHORT-LIVED EVENTS

As discussed in Section 3.2, event-based ICs and EALs define a variety of specific occurrences that have potential or actual safety significance. By their nature, some of these events may be short-lived and, thus, over before the emergency classification assessment can be completed. If an event occurs that meets or exceeds an EAL, the associated ECL must be declared regardless of its continued presence at the time of declaration. Examples of such events include a failure of the reactor protection system to automatically scram/trip the reactor followed by a successful manual scram/trip or an earthquake.

5.8 CLASSIFICATION OF TRANSIENT CONDITIONS

Many of the ICs and/or EALs contained in this document employ time-based criteria. These criteria will require that the IC/EAL conditions be present for a defined period of time before an emergency declaration is warranted. In cases where no time-based criterion is specified, it is recognized that some transient conditions may cause an EAL to be met for a brief period of time (e.g., a few seconds to a few minutes). The following guidance should be applied to the classification of these conditions.

<u>EAL momentarily met during expected plant response</u> - In instances where an EAL is briefly met during an expected (normal) plant response, an emergency declaration is not warranted provided that associated systems and components are operating as expected, and operator actions are performed in accordance with procedures.

<u>EAL momentarily met but the condition is corrected prior to an emergency declaration</u> – If an operator takes prompt manual action to address a condition, and the action is successful in correcting the condition prior to the emergency declaration, then the applicable EAL is not considered met and the associated emergency declaration is not required. For illustrative purposes, consider the following example.

An ATWS occurs and the auxiliary feedwater system fails to automatically start. Steam generator levels rapidly lower and the plant enters an inadequate RCS heat removal condition (a potential loss of both the fuel clad and RCS barriers). If an operator manually starts the auxiliary feedwater system in accordance with an EOP step and clears the inadequate RCS heat removal condition prior to an emergency declaration, then the classification should be based on the ATWS only.

It is important to stress that the 15-minute emergency classification assessment period is not a "grace period" during which a classification may be delayed to allow the performance of a corrective action that would obviate the need to classify the event; emergency classification assessments must be deliberate and timely, with no undue delays. The provision discussed above addresses only those rapidly evolving situations where an operator is able to take a successful corrective action prior to the Emergency Director completing the review and steps necessary to make the emergency declaration. This provision is included to ensure that any public protective actions resulting from the emergency classification are truly warranted by the plant conditions.

5.9 AFTER-THE-FACT DISCOVERY OF AN EMERGENCY EVENT OR CONDITION

In some cases, an EAL may be met but the emergency classification was not made at the time of the event or condition. This situation can occur when personnel discover that an event or condition existed which met an EAL, but no emergency was declared, and the event or condition no longer exists at the time of discovery. This may be due to the event or condition not being recognized at the time or an error that was made in the emergency classification process.

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR § 50.72 within one hour of the discovery of the undeclared event or condition. The licensee should also notify appropriate State and local agencies in accordance with the agreed upon arrangements.

5.10 RETRACTION OF AN EMERGENCY DECLARATION

Guidance on the retraction of an emergency declaration reported to the NRC is discussed in NUREG-1022.

6 ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT ICS/EALS

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

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
RU1 Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer. *Op. Modes: All*	**RA1** Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem THYROID CDE. *Op. Modes: All*	**RS1** Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem THYROID CDE. *Op. Modes: All*	**RG1** Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem THYROID CDE. *Op. Modes: All*
RU2 UNPLANNED loss of water level above irradiated fuel. *Op. Modes: All*	**RA2** Significant lowering of water level above, or damage to, irradiated fuel. *Op. Modes: All*	**RS2** Spent fuel pool level at 40'-4" or lower. *Op. Modes: All*	**RG2** Spent fuel pool level cannot be restored to at least 40'-4" for 60 minutes or longer. *Op. Modes: All*
	RA3 Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown. *Op. Modes: All*		

ECL: UNUSUAL EVENT

Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer.

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2 or 3)

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** of the following radiation monitor greater than the values listed in Table R1 column "UE" for 60 minutes or longer:

Table R1: Effluent Monitors					
Release Point	Monitor	GE	SAE	ALERT	UE
Unit Vent	RT-8010B	1.50 E+08 μCi/sec	1.50 E+07 μCi/sec	1.50 E+06 μCi/sec	1.40 E+05 μCi/sec
Main Steam Lines	RT-8046 thru 8049	4.00 E+02 μCi/cm^3	4.00 E+01 μCi/cm^3	4.00 E+00 μCi/cm^3	5.00 E-02 μCi/cm^3

(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.

(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.

Basis:

This IC addresses a potential lowering 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.

STPEGS incorporated 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 normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways.

EAL #2- 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 #3- 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 RA1.

RU1: EAL-1 Selection Basis

The Unit Vent and Main Steam Line monitor readings were included in this EAL because they give instantaneous indications of a monitored gaseous release exceeding twice the ODCM limits. Normal gaseous effluents are due to planned RCB purges and monitored by the Unit Vent. The Main Steam Line monitor readings were included because they correspond to a concentration that would result in a release rate of twice the ODCM limits if there were a release via the Power Operated Relief Valves (PORVs) or Safety Relief Valves. A release from the PORVs or Safety Relief Valves is not a normal effluent pathway but meets the intent of the EAL.

The Unit Vent and Main Steam Line release values are based on Calculation No. STPNOC013-CALC-002, Rev. 2.

RU1: EAL-2, 3 Selection Basis

For EAL-2, there are two effluent radiation monitors, RT-8038 (liquid) and RT-8010B (gaseous), however only RT-8010B was included. The alarm setpoint for the gaseous effluent radiation monitor RT-8010B is set at the ODCM limits. An indication of two times the alarm setpoint (two times the ODCM limit) would allow operators time to secure the release prior to meeting this EAL. The liquid effluent radiation monitor RT-8038 was not included in EAL-2 because the activity in liquid discharges is normally the several orders of magnitude lower than the ODCM limits. In order to alert personnel to significant changes in the liquid effluent activity, the alarm setpoint for RT-8038 is normally set several orders of magnitude below the ODCM limits. Setting the alarm setpoint for RT-8038 at the ODCM limit would remove this capability and violate the intent of the EAL.

For EAL-3, sample analysis could be used as a backup for the effluent monitor indications.

REFERENCES:

1. Calculation No: STPNOC013-CALC-002 Rev. 2, Radiological Release Thresholds for Emergency Action Levels
2. Offsite Dose Calculation Manual (ODCM), Rev. 17, Part B3.0 to B4.9
3. UFSAR, Rev. 14, Section 11.5.2.3.3 and 11.5.2.5.3 (monitor descriptions)
4. UFSAR, Rev. 14, Section 11.5.2.4.4 (liquid waste processing monitor)

ECL: UNUSUAL EVENT

Initiating Condition: UNPLANNED loss of water level above irradiated fuel.

Operating Mode Applicability: ALL

Emergency Action Level:

(1) a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following:

- Visual Observation

  OR

- Annunciator alarm on lampbox 22M02 Window F-5 "SFP WATER LVL HI/LO"

  OR

- Spent fuel in the ICSA **AND** Annunciator alarm on lampbox 22M02 Window F-6 "SFP Trouble" **AND** Plant Computer point FCLC1420 "REFLNG CAV LVL IN CNTMT" (ICSA Water Level HI/LO) is in alarm

AND

b. UNPLANNED rise in area radiation levels on ANY of the following radiation monitors.

- RE-8055 (68' RCB) - Mode 5 or 6 only

  OR

- RE-8099 (68' RCB) - Mode 5 or 6 only

  OR

- RE-8090 (68' FHB)

Basis:

This IC addresses a lowering in water level above irradiated fuel sufficient to cause elevated radiation levels. This condition could be a precursor to a more serious event and is also indicative of a minor loss in the ability to control radiation levels within the plant. It is therefore a potential degradation in the level of safety of the plant.

A water level lowering will be primarily determined by indications from available level instrumentation. Other sources of level indications may include reports from plant personnel (e.g., from a refueling crew) or video camera observations. A significant drop in the water level may also cause a rise in the radiation levels of adjacent areas that can be detected by monitors in those locations.

The effects of planned evolutions should be considered. For example, a refueling bridge area radiation monitor reading may rise due to planned evolutions such as lifting of the reactor vessel head or movement of a fuel assembly. Note that this EAL is applicable only in cases where the elevated reading is due to an UNPLANNED loss of water level.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC RA2.

RU2: EAL-1 Selection Basis

Hi/Lo level sensors are located in the Spent Fuel Pool (LSHL 1401) and the RCB, In Containment Storage Area (ICSA) (LSHL 1420). If level in the Spent Fuel Pool rises or lowers by more than 6 inches above or below the normal water level of 66'-6" (UFSAR 9.1.2.1), the "SFP WATER LEVEL HI/LO" lampbox 22M02 window F-5 annunciator alarm is received in the Control Room (0POP09-AN-22M2, Annunciator Lampbox 22M02 Response Instructions).

Although the ICSA has a Hi/LO level sensor, there is not an annunciator in the Control Room similar to the one for the Spent Fuel Pool. There is however, a "SFP TROUBLE" lampbox 22M02 window F-6 annunciator in the control room. One of the inputs to this alarm is FC-LSHL-1420, the ICSA HI/LO level sensor. Since no fuel is located in the ICSA in modes 1-4, this EAL only applies in modes 5 or 6.

Area radiation monitors RE-8055 and RE-8099 are located are located in the RCB 68' elevation on the bioshield wall close to the refueling cavity. Area radiation monitor RE-8090 is located in the Fuel Handling Building on 68' Elevation near the Spent Fuel Pool.

Expected radiation levels for a loss of water level can range from a few mR/hr to thousands of R/hr.
For a drop of water level of approximately 14' (from 66'-6" to 51'-10") with approximately 13' of water over the top of any array, the dose rate would be expected not to exceed 2.5 mR/hr, above background. This assumes 42 hours of decay with a full core off-load (section 9 of STP UFSAR).

For a significant drop of water level that would still cover the arrays, the radiation levels could range from several hundred R/hr to over a thousand R/hr on and around the 68' elevation deck (table C-5 NUREG CR/0649).

REFERENCES:

1. 0POP09-AN-22M2, Rev. 25, Annunciator Lampbox 22M02 Response Instructions F-5 and F-6 Window (level alarms)
2. 0POP04-FC-0001, Rev. 29, Loss of Spent Fuel Pool Level or Cooling (level alarms)
3. Technical Specification, amendment 104 (Unit 1) and 91 (Unit 2), Section 5.6.2 (Design water level)
4. UFSAR, Rev. 16, Section 9.1.2.1 (Dose rates)
5. UFSAR, Rev. 16, Section 9.1.2.2 (Normal water level)
6. NUREG CR/0649 (Dose rates), reference only (not included in submittal)
7. Drawing 5R219F05028#1 Spent Fuel Pool Cooling and Cleanup System (level sensors)
8. UFSAR, Rev. 15, table 12.3.4-1, Area Radiation Monitors

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: ALL

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 values listed in Table R1 column "ALERT" for 15 minutes or longer:

Table R1: Effluent Monitors

Release Point	Monitor	GE	SAE	ALERT	UE
Unit Vent	RT-8010B	1.50 E+08 μCi/sec	1.50 E+07 μCi/sec	1.50 E+06 μCi/sec	1.40 E+05 μCi/sec
Main Steam Lines	RT-8046 thru 8049	4.00 E+02 μCi/cm^3	4.00 E+01 μCi/cm^3	4.00 E+00 μCi/cm^3	5.00 E-02 μCi/cm^3

(2) Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem THYROID CDE at or beyond the SITE BOUNDARY.

(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.

(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.

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.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC RS1.

RA1: EAL-1 Selection Basis

The Unit Vent and Main Steam Line monitor readings were included in this EAL because they give instantaneous indications of a monitored gaseous release meeting the EAL threshold values of 10 mrem TEDE or 50 mrem CDE THYROID at the SITE BOUNDARY. Gaseous releases from the plant are monitored by the Unit Vent. The Main Steam Line monitor readings correspond to a concentration that would result in a release rate meeting the EAL threshold values if there were a release via the Power Operated Relief Valves (PORVs) or Safety Relief Valves.

The Unit Vent and Main Steam Line release values are based on Calculation No. STPNOC013-CALC-002, Rev. 2. The adjusted values used in this EAL were conservatively truncated by less than 1% of the calculated values to ensure they are readily assessable.

RA1: EAL-2, 3, 4 Selection Basis

N/A

REFERENCES:

1. Calculation No: STPNOC013-CALC-002 Rev. 2., Radiological Release Thresholds for Emergency Action Levels
2. UFSAR, Rev. 14, Section 11.5.2.3.3 and 11.5.2.5.3 (monitor descriptions)
3. UFSAR, Rev. 14, 11.5.2.4.4 (liquid waste processing monitor)

RA2

ECL: ALERT

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel.

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2 or 3)

(1) Uncovery of irradiated fuel in the REFUELING PATHWAY.

(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:

- FHB Exhaust, RT-8035 or RT-8036 greater than 1.00 E-1 $\mu Ci/cm^3$

  OR

- ARM (68' FHB), RE-8090 greater than 1,500 mR/hr

OR

b. Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by **ANY** of the following RCB radiation monitor readings (Mode 5 or 6 only).

- ARMs (68' RCB), RE-8055 or RE-8099 greater than 850 mR/hr.

NOTE
EAL-3 is not applicable until the enhanced SFP level instrumentation is available for use.

(3) Lowering of spent fuel pool level to 49'-10" or lower.

Basis:

This IC addresses events that have caused IMMINENT or actual damage to an irradiated fuel assembly, *or a significant lowering of water level within the spent fuel pool or Inside Containment Storage Area (ICSA)*.These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC E-HU1.

Escalation of the emergency would be based on either Recognition Category R or C ICs.

EAL #1- This EAL escalates from RU2 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used (e.g., a boil-off curve). Classification of an event using this EAL should be based on the totality of available indications, reports and observations. While an area radiation monitor could detect a rise in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.
A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

EAL #2- This EAL addresses a release of radioactive material caused by mechanical damage to irradiated fuel. Damaging events may include the dropping, bumping or binding of an assembly, or dropping a heavy load onto an assembly. A rise in readings on radiation monitors should be considered in conjunction with in-plant reports or observations of a potential fuel damaging event (e.g., a fuel handling accident).

EAL #3- Spent fuel pool water level at this value is within the lower end of the level range necessary to prevent significant dose consequences from direct gamma radiation to personnel performing operations in the vicinity of the spent fuel pool. This condition reflects a significant loss of spent fuel pool water inventory and thus it is also a precursor to a loss of the ability to adequately cool the irradiated fuel assembles stored in the pool.

Escalation of the EMERGENCY CLASSISFICATION LEVEL would be via ICs RS1 or RS2.

RA2: EAL-2 Selection Basis:

The calculated airborne source term and radiation monitor responses for a fuel handling accident in the FHB is based on Calculation STPNOC013-CALC-005 Rev.2. The threshold value of 1500 mR/hr for area radiation monitor RE-8090 was truncated less than 4% from the calculated value to ensure the threshold was readily assessable. Threshold values for FHB Exhaust Monitors RT-8035 and RT-8036 were also included because they are accident monitors that are sensitive to noble gases which are expected to be present if irradiated fuel is damaged. The calculated monitor reading for RT-8035 and RT-8036 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.

The calculated airborne source term and radiation monitor response for a fuel handling accident in the RCB is based on Calculation STPNOC013-CALC-005 Rev.2. The threshold value of 850 mR/hr for RE-8055 and RE-8099 was truncated less than 2% from the calculated value to ensure the threshold is readily assessable.

RA2: EAL-3 Selection Basis:

Spent Fuel Pool level of 49'- 10" (Level 2) is a site specific level based on the guidance provided in NEI 12-02, Revision 1, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licensees with Regard to Reliable Spent Fuel Pool Instrumentation", August 2012.

In NRC Order EA-12-051 and NEI 12-02, Level 2 is defined as the "*level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck ...* "

The STP UFSAR identifies the top of the Spent Fuel Storage Racks at 39'- 10". The guidance in NEI 12-02 indicates that 10' of water above the top of the Spent Fuel Storage Racks provides substantial radiation shielding. Ten feet of water above the Spent Fuel Storage Racks is 49'- 10", the threshold value for this EAL.

Reference 6 identifies the site specific levels of the proposed SFP level instrument and identifies the Level 2 criteria as 49'– 10".

REFERENCES:

1. Calculation No.: STPNOC013-CALC-005 Rev.2, Fuel Handling Accident Monitor Response for Emergency Action Levels.
2. UFSAR, Rev. 16, Section 9.1.2.1 (SFP Rad levels)
3. UFSAR, Rev. 16, Section 9.1.2.2 (SFP top of Racks)
4. NRC Order EA-12-051 (SFP levels)
5. NEI 12-02, Rev. 1 (SFP levels)
6. South Texas Project (STP) Overall Integrated Plan for Implementation of Unit 1 & Unit 2 Spent Fuel Pool Level Instrumentation to Meet NRC Order EA-12-051, Rev. 0, NOC –AE-13002959

ECL: ALERT

Initiating Condition: Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2)

Note: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

(1) Dose rate greater than 15 mR/hr in **ANY** of the following areas:

- Control Room ARM (RE-8066)

  OR

- Central Alarm Station (CAS) by radiation survey

(2) An UNPLANNED event results in radiation levels that prohibit or impede access to **ANY** of the areas listed in Table H3/R2:

TABLE H3/R2: Plant Areas Requiring Access		
MODE 4	RCB	RHR Heat Exchanger Rooms
	MAB	51 ft Room 335
	EAB	Roof, MCC 1G8, 4.16KV Switchgear Rooms
MODE 5	EAB	4.16KV Switchgear Rooms

Basis:

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Emergency Director should consider the cause of the higher radiation levels and determine if another IC may be applicable.

For EAL #2, an ALERT declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the higher radiation levels. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., installing temporary shielding, requiring use of non-routine protective equipment, requesting an extension in dose limits beyond normal administrative limits).

An emergency declaration is not warranted if any of the following conditions apply.

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the elevated radiation levels). For example, the plant is in Mode 1 when the radiation rise occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The higher radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via Recognition Category R, C or F ICs.

RA3: EAL-1, EAL-2 Selection Basis:

The NEI 99-01 value of 15 mR/hr is derived from the GDC 19 value of 5 rem in 30 days with adjustment for expected occupancy times. The rooms listed in EAL-1 require continuous occupancy to maintain normal plant operation, or to perform a normal cooldown or shutdown.

The areas listed in EAL-2 apply to areas that contain equipment necessary for plant operations, cooldown, or shutdown. Assuming all plant equipment is operating as designed, Normal operations and safe shutdown equipment operation is capable from the Main Control Room (MCR). The plant is able to transition into a hot shutdown from the MCR, therefore H3/R2 is a list of plant rooms or areas with entry-related mode applicability that contain equipment which require a manual/local action necessary following entry into hot shutdown (establish Residual Heat Removal shutdown cooling, disable operation of charging and ECCS equipment, and limit dilution pathways) and subsequent entry into cold shutdown (disable operation of ECCS equipment). After achieving cold shutdown it is assumed that the plant will be maintained in a cold shutdown condition.

REFERENCES:

1. General Design Criteria 19
2. 0POP03-ZG-0008, Rev. 56, Power Operations
3. 0POP03-ZG-0006, Rev. 54, Plant Shutdown from 100% to Hot Standby
4. 0POP03-ZG-0007, Rev. 71, Plant Cooldown

RS1

ECL: SITE AREA EMERGENCY

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem THYROID CDE.

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2 or 3)

Notes:

- The Emergency Director should declare the SITE AREA EMERGENCY 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 values listed in Table R1 column "SAE" for 15 minutes or longer:

Table R1: Effluent Monitors					
Release Point	**Monitor**	**GE**	**SAE**	**ALERT**	**UE**
Unit Vent	RT-8010B	1.50 E+08 μCi/sec	1.50 E+07 μCi/sec	1.50 E+06 μCi/sec	1.40 E+05 μCi/sec
Main Steam Lines	RT-8046 thru 8049	4.00 E+02 μCi/cm^3	4.00 E+01 μCi/cm^3	4.00 E+00 μCi/cm^3	5.00 E-02 μCi/cm^3

(2) Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem THYROID CDE at or beyond the SITE BOUNDARY.

(3) 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.

Basis:

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA PROTECTIVE ACTION GUIDES (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

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 10% of the EPA PAG of 1,000 mrem while the 500 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.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC RG1.

RS1: EAL-1 Selection Basis:

The Unit Vent and Main Steam Line monitor readings were included in this EAL because they give instantaneous indications of a monitored gaseous release meeting the EAL threshold values of 100 mrem TEDE or 500 mrem CDE THYROID at the SITE BOUNDARY. Gaseous releases from the plant are monitored by the Unit Vent. The Main Steam Line monitor readings correspond to a concentration that would result in a release rate meeting the EAL threshold values if there were a release via the Power Operated Relief Valves (PORVs) or Safety Relief Valves.

The Unit Vent and Main Steam Line release values are based on Calculation No. STPNOC013-CALC-002 Rev.2. The adjusted values used in this EAL were conservatively truncated by less than 1% of the calculated values to ensure they are readily assessable.

RS1: EAL-2, EAL-3 Selection Basis:

N/A

REFERENCES:

1. Calculation No: STPNOC013-CALC-002 Rev.2, Radiological Release Thresholds for Emergency Action Levels
2. UFSAR Section, Rev. 14, Section 11.5.2.3.3 and 11.5.2.5.3 (monitor descriptions)

RS2

ECL: SITE AREA EMERGENCY

Initiating Condition: Spent fuel pool level at 40'-4" or lower.

Operating Mode Applicability: ALL

Emergency Action Level:

NOTE
EAL-1 is not applicable until the enhanced SFP level instrumentation is available for use.

(1) Lowering of spent fuel pool level to 40'-4" or lower.

Basis:

This IC addresses a significant loss of spent fuel pool inventory control and makeup capability leading to IMMINENT fuel damage. This condition entails major failures of plant functions needed for protection of the public and thus warrant a SITE AREA EMERGENCY declaration.

It is recognized that this IC would likely not be met until well after another SITE AREA EMERGENCY IC was met; however, it is included to provide classification diversity.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC RG1 or RG2.

RS2: EAL-1 Selection Basis:

Spent Fuel Pool level of 40'- 4" (Level 3) is a site specific level based on the guidance provided in NEI 12-02, Revision 1, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation", August 2012.

In NRC Order EA-12-051 and NEI 12-02, Level 3 is defined as "*level where fuel remains covered and actions to implement make-up water addition should no longer be deferred*."

The STP UFSAR identifies the top of the Spent Fuel Storage Racks at 39'- 10".

Reference 4 identifies the site specific levels for the proposed SFP level instrumentation and identifies the Level 3 criteria as 40'– 4".

REFERENCES:

1. UFSAR, Rev. 16, Section 9.1.2.2 (SFP top of Racks)
2. NRC Order EA-12-051 (SFP Levels)
3. NEI 12-02, Revision 1, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation", August 2012
1. South Texas Project (STP) Overall Integrated Plan for Implementation of Unit 1 & Unit 2 Spent Fuel Pool Level Instrumentation to Meet NRC Order EA-12-051, Rev. 0, NOC –AE-13002959

ECL: GENERAL EMERGENCY

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem THYROID CDE.

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2 or 3)

Notes:

- The Emergency Director should declare the GENERAL EMERGENCY 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 values listed in Table R1 column "GE" for 15 minutes or longer:

Table R1: Effluent Monitors

Release Point	Monitor	GE	SAE	ALERT	UE
Unit Vent	RT-8010B	1.50 E+08 μCi/sec	1.50 E+07 μCi/sec	1.50 E+06 μCi/sec	1.40 E+05 μCi/sec
Main Steam Lines	RT-8046 thru 8049	4.00 E+02 μCi/cm^3	4.00 E+01 μCi/cm^3	4.00 E+00 μCi/cm^3	5.00 E-02 μCi/cm^3

(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.

(3) Field survey results indicate **EITHER** of the following at or the SITE BOUNDARY:

- Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer.

  OR

- Analyses of field survey samples indicate THYROID CDE greater than 5,000 mrem for one hour of inhalation.

Basis:

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA PROTECTIVE ACTION GUIDES (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

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 the EPA PAG of 1,000 mrem while the 5,000 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.

RG1: EAL-1 Selection Basis:

The Unit Vent and Main Steam Line monitor readings were included in this EAL because they give instantaneous indications of a monitored gaseous release meeting the EAL threshold values of 1000 mrem TEDE or 5000 mrem CDE THYROID at the SITE BOUNDARY. Gaseous releases from the plant are monitored by the Unit Vent. The Main Steam Line monitor readings correspond to a concentration that would result in a release rate meeting the EAL threshold values if the release was via the Power Operated Relief Valves (PORVs) or Safety Relief Valves.

The Unit Vent and Main Steam Line release values are based on Calculation No. STPNOC013-CALC-002 Rev.2. The adjusted values used in this EAL were conservatively truncated by less than 1% of the calculated values to ensure they are readily assessable.

RG1: EAL-2, EAL-3 Selection Basis:

N/A

REFERENCES:

1. Calculation No: STPNOC013-CALC-002 Rev.2, Radiological Release Thresholds for Emergency Action Levels,
2. STP UFSAR, Rev. 14, Section 11.5.2.3.3 and 11.5.2.5.3 (monitor descriptions)

ECL: GENERAL EMERGENCY

Initiating Condition: Spent fuel pool level cannot be restored to at least 40'-4" for 60 minutes or longer.

Operating Mode Applicability: ALL

Emergency Action Level:

Note: The Emergency Director should declare the GENERAL EMERGENCY promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.

NOTE
EAL-1 is not applicable until the enhanced SFP level instrumentation is available for use.

(1) Spent fuel pool level cannot be restored to at least40'-4" for 60 minutes or longer.

Basis:

This IC addresses a significant loss of spent fuel pool inventory control and makeup capability leading to a prolonged uncovery of spent fuel. This condition will lead to fuel damage and a radiological release to the environment.

It is recognized that this IC would likely not be met until well after another GENERAL EMERGENCY IC was met; however, it is included to provide classification diversity.

RG2: EAL-1 Selection Basis:

The Spent Fuel Pool level of 40'- 4" (Level 3) is a site specific level based on the guidance provided in NEI 12-02, Revision 1, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation", August 2012.

In NRC Order EA-12-051 and NEI 12-02, Level 3 is defined as *"level where fuel remains covered and actions to implement make-up water addition should no longer be deferred."*

The STP UFSAR identifies the top of the Spent Fuel Pool Racks at 39'- 10".

Reference 4 identifies the site specific levels of the proposed level instrumentation and identifies the Level 3 criteria as 40'- 4".

REFERENCES:

1. UFSAR, Rev. 16, Section 9.1.2.2 (SFP top of Racks)
2. NRC Order EA-12-051 (SFP Levels)
3. NEI 12-02, Rev. 1, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation", August 2012
4. South Texas Project (STP) Overall Integrated Plan for Implementation of Unit 1 & Unit 2 Spent Fuel Pool Level Instrumentation to Meet NRC Order EA-12-051, Rev. 0, NOC –AE-13002959

7 COLD SHUTDOWN / REFUELING SYSTEM MALFUNCTION ICS/EALS

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

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
CU1 UNPLANNED loss of RCS inventory for 15 minutes or longer. *Op. Modes:* 5,6	**CA1** Loss of RCS inventory. *Op. Modes:* 5,6	**CS1** Loss of RCS inventory affecting core decay heat removal capability. *Op. Modes:* 5,6	**CG1** Loss of RCS inventory affecting fuel clad integrity with containment challenged. *Op. Modes:* 5,6
CU2 Loss of **ALL** but one AC power source to emergency buses for 15 minutes or longer. *Op. Modes:* 5,6 *Defueled*	**CA2** Loss of **ALL** offsite and **ALL** onsite AC power to emergency buses for 15 minutes or longer. *Op. Modes:* 5,6 , *Defueled*		
CU3 UNPLANNED rise in RCS temperature. *Op. Modes:* 5,6	**CA3** Inability to maintain the plant in cold shutdown. *Op. Modes:* 5,6		
CU4 Loss of Vital DC power for 15 minutes or longer. *Op. Modes:* 5,6			
CU5 Loss of **ALL** onsite or offsite communications capabilities. *Op. Modes:* 5,6, *Defueled*			
	CA6 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. *Op. Modes:* 5,6		

CU1

ECL: UNUSUAL EVENT

Initiating Condition: UNPLANNED loss of RCS inventory for 15 minutes or longer.

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2)

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

(1) UNPLANNED loss of reactor coolant results in RCS level below the procedurally required limit for 15 minutes or longer.

(2) a. RCS level cannot be monitored.

AND

b. UNPLANNED rise in **ANY** of the following sump or tank levels in Table C2:

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

Basis:

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor RCS level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that lower RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an UNUSUAL EVENT due to the reduced water inventory that is available to keep the core covered.

EAL #1- recognizes that the minimum required RCS level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is specified in the applicable STP operating procedure.

The 15-minute threshold duration allows sufficient time for prompt operator actions to restore and maintain the expected water level. This criterion excludes transient conditions causing a brief lowering of water level.

EAL #2- addresses a condition where all means to determine RCS level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS

Continued loss of RCS inventory may result in escalation to the ALERT EMERGENCY CLASSIFICATION LEVEL via either IC CA1 or CA3.

CU1 – EAL-1 Selection Basis:

RCS inventory is maintained above the reactor vessel flange (39'-3") during refueling outages per 0POP03-ZG-0007, Plant Cooldown. RCS level may be lowered below the vessel flange for specific purposes (e.g., head removal, mid-loop operations) as described in 0POP03-ZG-0009, Mid-Loop Operation. The 15 minute time frame allows for prompt operator actions to restore RCS level in the event of an UNPLANNED lowering of RCS level below the prescribed operating limit.

CU1 – EAL-2 Selection Basis:

This EAL includes two conditions. The first condition is the inability to monitor RCS level and the second condition provides secondary indications that inventory loss may be occurring.

The secondary indicators of inventory loss include a list of tanks/sumps found in 0POP04-RC-0003, Excessive RCS Leakage. Since other system leaks could rise levels in various tanks and sumps, the list has been limited to the tanks and sumps that would have the highest probability of indicating RCS leakage inside the Reactor Containment Building.

Although procedure 0POP04-RC-0003 is designated for use in modes 1-4, its logic is applicable to this EAL.

REFERENCES:

1. 0POP04-RC-0003, Rev. 18, Excessive RCS Leakage
2. 0POP03-ZG-0007, Rev. 71, Plant Cooldown
3. 0POP03-ZG-0009, Rev. 59, Mid-Loop Operation

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: 5, 6, Defueled

Emergency Action Level:

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

(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.

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.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as an ALERT because of the additional time available to restore another power source to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an emergency bus. 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 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 fed from an onsite or offsite power source.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.
The subsequent loss of the remaining single power source would escalate the event to an ALERT in accordance with IC CA2.

CU2: EAL-1 Selection Criteria:

The condition indicated by this EAL is the degradation of the offsite and onsite power systems such that any additional single failure would results in a loss of all AC power. This condition is an UNUSUAL EVENT during modes 5, 6 and Defueled because of the additional time available to restore power due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. In modes 1-4, this condition is an ALERT as described in SA1.

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

ECL: UNUSUAL EVENT

Initiating Condition: UNPLANNED rise in RCS temperature.

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2)

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

(1) UNPLANNED rise in RCS temperature to greater than 200 °F (Tavg).

(2) Loss of **ALL** RCS temperature and RCS level indication for 15 minutes or longer.

Basis:

This IC addresses an UNPLANNED rise in RCS temperature above the Technical Specification cold shutdown temperature limit, or the inability to determine RCS temperature and level, represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Director should also refer to IC CA3.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

EAL #1- involves a loss of decay heat removal capability, or an addition of heat to the RCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

During an outage, the level in the reactor vessel will normally be maintained above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid rise in reactor coolant temperature depending on the time after shutdown.

EAL #2- reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication. Escalation to ALERT would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

CU3: EAL-1 Selection Basis:

An UNPLANNED temperature rise above 200 °F would result in an UNPLANNED mode change due to the inability to control RCS temperature. Mode 4 (Hot Shutdown) would be entered when Tavg exceeds 200 °F (Reference 1).

CU3: EAL-2 Selection Basis:

N/A

REFERENCES:

1. Technical Specifications Table 1.2 (Mode, Temperature, Power, k_{eff} Table)

CU4

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: 5, 6

Emergency Action Level:

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

(1) Indicated voltage is less than 105.5 VDC on required Vital DC buses for 15 minutes or longer.

Basis:

This IC addresses a loss of Vital DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or refueling mode. In these modes, the core decay heat load has been significantly reduced, and coolant system temperatures and pressures are lower; these conditions extend the time available to restore a vital DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, "required" means the Vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if Train A and C are out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an UNUSUAL EVENT. A loss of Vital DC power to Train A and/or C would not warrant an emergency classification.

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

Depending upon the event, escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC CA1 or CA3, or an IC in Recognition Category R.

CU4 – EAL-1 Selection Basis:

The minimum voltage for Class 1E 125 VDC battery buses was determined in calculation 13-DJ-006, Rev. 3 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. Calculation 13-DJ-006, Rev. 0, 125 VDC Battery Four Hour Coping Analysis
2. 0POP05-EO-EC00, Rev. 23, Loss of All AC Power

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: 5, 6, Defueled

Emergency Action Levels: (1 or 2 or 3)

(1) Loss of **ALL** of the following Onsite communication methods in Table C4.

(2) Loss of **ALL** of the following Offsite Response Organization (ORO) communication methods in Table C4.

(3) Loss of **ALL** of the following NRC communication methods in Table C4.

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

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 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.

CU5: EAL-1, EAL-2, and 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, Rev. 8, Emergency Communications

CA1

ECL: ALERT

Initiating Condition: Loss of RCS inventory.

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2)

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

(1) Loss of RCS inventory as indicated by level less than 32 ft. 9 inch (+ 6 inches above hot leg centerline).

(2) a. RCS level cannot be monitored for 15 minutes or longer

AND

b. UNPLANNED rise in **ANY** of the following sump or tank levels in Table C2 due to a loss of reactor vessel/RCS inventory.

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

Basis:

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

EAL #1- A lowering of water level below elevation 32'- 9" indicates that operator actions have not been successful in restoring and maintaining reactor vessel/ water level. The heat-up rate of the coolant will rise as the available water inventory is reduced. A continuing reduction in water level will lead to core uncovery. Although related, EAL #1 is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Residual Heat Removal suction point). Arise in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

EAL #2- The inability to monitor reactor vessel/RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to

ensure they are indicative of leakage from the reactor vessel/RCS. The 15-minute duration for the loss of level indication was chosen because it is half of the EAL duration specified in IC CS1

If the reactor vessel/RCS inventory level continues to lower, then escalation to SITE AREA EMERGENCY would be via IC CS1.

CA1: EAL-1 Selection Basis:

The minimum RCS level at which an RHR pump can be started per 0POP02-RH-0001 is 32 feet 9 inches (+ 6 inches above hot leg centerline). If RCS inventory is reduced below this level, normal decay heat removal systems may not be available for core cooling. This threshold is not applicable to reduced inventory vacuum fill since this is a controlled evolution and not indicative of an RCS loss.

CA1: EAL-2 Selection Basis:

The tanks/sumps selected for this EAL were obtained from 0POP04-RC-0003, Excessive RCS Leakage. Since other system leaks could raise levels in various tanks and sumps, the list was limited to the tanks and sumps that would have the highest probability of indicating RCS leakage inside the Reactor Containment Building.

Although procedure 0POP04-RC-0003 is designated for use in modes 1-4, its logic is applicable to this EAL.

REFERENCES:

1. 0POP04-RC-0003, Rev. 18, Excessive RCS Leakage
2. 0POP02-RH-0001, Rev. 63, Residual Heat Removal System Operation

CA2

ECL: ALERT

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

Operating Mode Applicability: 5, 6, Defueled

Emergency Action Level:

Note: The Emergency Director should declare the ALERT 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 **ALL** three 4160V AC ESF Busses 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.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as a SITE AREA EMERGENCY because of the additional time available to restore an emergency bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition represents an actual or potential substantial degradation of the level of safety of the plant.

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

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC CS1 or RS1.

CA2 – 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

ECL: ALERT

Initiating Condition: Inability to maintain the plant in cold shutdown.

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2)

Note: The Emergency Director should declare the ALERT promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

(1) UNPLANNED rise in RCS temperature to greater than 200 °F (Tavg) for greater than the duration specified in Table C3.

Table C3: RCS Heat-up Duration Thresholds		
RCS Status	**Containment Closure Status**	**Heat-up Duration**
Intact (but not at reduced inventory)	Not applicable	60 minutes*
Not intact (or at reduced inventory)	Established	20 minutes*
	Not Established	0 minutes
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

(2) UNPLANNED RCS pressure rise greater than 10 psig. (This EAL does not apply during water-solid plant conditions.)

Basis: This IC addresses conditions involving a loss of decay heat removal capability or an addition of heat to the RCS in excess of that which can currently be removed. Either condition represents an actual or potential substantial degradation of the level of safety of the plant.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

EAL #1-The RCS Heat-up Duration Thresholds table addresses an rise in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not intact, or RCS inventory is reduced (e.g., mid-loop operation). The 20-minute criterion was included to allow time for operator action to address the temperature rise.

The RCS Heat-up Duration Thresholds table also addresses an rise in RCS temperature with the RCS intact. The status of CONTAINMENT CLOSURE is not crucial in this condition since the intact RCS is providing a high pressure barrier to a fission product release. The 60-minute time frame should allow sufficient time to address the temperature rise without a substantial degradation in plant safety.

Finally, in the case where there is a rise in RCS temperature, the RCS is not intact or is at reduced inventory and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because 1) the evaporated reactor coolant may be released directly into the Containment atmosphere and subsequently to the environment, and 2) there is reduced reactor coolant inventory above the top of irradiated fuel.

EAL #2- provides a pressure-based indication of RCS heat-up.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC CS1 or RS1.

CA3 – EAL-1 Selection Basis:

Table C3 was adopted from NEI 99-01, Rev. 6. This EAL addresses the concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal. A number of phenomena such as pressurization, vortexing, steam generator U-tube draining, RCS level differences when operating at a mid-loop condition, decay heat removal system design, and level instrumentation problems can lead to conditions where decay heat removal is lost and core uncover can occur. NRC analyses show that there are sequences that can cause core uncovery in 15 to 20 minutes, and severe core damage within an hour after decay heat removal is lost. The allowed time frames are consistent with the guidance provided by Generic Letter 88-17 and believed to be conservative given that a low pressure containment barrier to fission product release is established.

CA3 – EAL-2 Selection Basis:

An UNPLANNED RCS pressure rise greater than 10 psig provides a pressure-based indication of RCS heat-up. The pressure change, per NEI 99-01 Rev. 6, is the lowest change in pressure that can be accurately determined using installed instrumentation, but not less than 10 psig.

REFERENCES:

1. Technical Specifications Table 1.2 (Mode, Temperature, Power, keff Table)

ECL: ALERT

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

Operating Mode Applicability: 5, 6

Emergency Action Level:

(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

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.

<u>Basis:</u>

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 would be via IC CS1 or RS1.

CA6: EAL-1 Selection Basis:

The listed hazards are taken directly from NEI 99-01, Rev. 6. The only additional hazard was the inclusion of the Main Cooling Reservoir since it is a credible hazard and analyzed in the STPEGS UFSAR (reference 2).

REFERENCES:

1. STPEGS UFSAR, Rev. 13, Section 3.4.1, Flood Protection

CS1

ECL: SITE AREA EMERGENCY

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

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2 or 3)

Note: The Emergency Director should declare the SITE AREA EMERGENCY promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.

(1) a. CONTAINMENT CLOSURE not established.

AND

b. RCS level less than 33% of plenum.

(2) a. CONTAINMENT CLOSURE established.

AND

b. RCS level less than 0% of plenum

(3) a. RCS level cannot be monitored for 30 minutes or longer.

AND

b. Core uncovery is indicated by **ANY** of the following:

- Reactor Containment Building, 68'-0" Area Radiation Monitors RE-8055 or RE-8099 reading greater than 9,000 mR/hr.

  OR

- Erratic source range monitor indication.

  OR

- 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

Basis:

This IC addresses a significant and prolonged loss of reactor vessel/RCS inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a SITE AREA EMERGENCY declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in RCS level. If RCS level cannot be restored, fuel damage is probable.

Outage/shutdown contingency plans provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS levels of EALs 1.b and 2.b reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

In EAL 3.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

These EALs address concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.*

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC CG1 orRG1.

CS1: EAL-1 Selection Basis:

Per NEI 99-01 Rev. 6, the RCS level indication should be six inches (6") below the bottom inside diameter of the RCS loop penetration at the reactor vessel. Six inches (6") below the bottom inside diameter of the RCS hot leg nozzle (elevation 31'-0.5") is elevation 30'-6.5" per 0POP03-ZG-0009, Mid-Loop Operation, Addendum 1, RCS/RHR Simplified Elevation Diagram. The nearest RVWL Monitoring System thermocouples are located 6 inches above (Sensor 6) and 4.9 inches below (Sensor7) the prescribed elevation of 30'-6.5". When water level is at the desired elevation of 30'-6.5", Sensor 6 will be dry and Sensor 7 will be wet. This condition corresponds to a reading of 33% of plenum per 0POP02- II-0002, RVWL Monitoring System, Addendum 1, RVWL Sensor Elevations.

CS1: EAL-2 Selection Basis:

Per NEI 99-01 Rev. 6, the RCS level indication should be approximately the top of active fuel (TAF). The RCS level which corresponds to the top of the active fuel is 26'-1" ~~28'-2" (0POP03-ZG-0009, Mid-Loop Operation, Addendum 1, RCS/RHR Simplified Elevation Diagram)~~. The nearest Reactor Vessel Water Level Monitoring System thermocouple to TAF is Sensor 8 at elevation 29'-2.7". Use of RVWL to approximate TAF; with the inherent gap of ~~12~~ 37 inches between indicated level and actual level, is acceptable for the purposes of signaling that the threat to the public is reduced when CONTAINMENT CLOSURE is established.

CS1: EAL-3 Selection Basis:

As RCS level drops the dose rates above the core will rise. Area Radiation Monitors RE-8055 and RE-8099 are located on the 68'-0" elevation of the reactor containment building. Their locations are identified on drawing 9C129A81105. Their range (0.1 mR/hr to 10,000 mR/hr) is identified in Table 12.3.4-1 of Section 12 of the UFSAR. A rising trend on these monitors can be an indication that core uncovery is occurring. Additionally, erratic source range monitor indications, or large level rises in the tanks listed can give further indication of core uncovery.

The threshold value for radiation monitors RE-8055 and RE-8099 was based on Calculation STPNOC013-CALC-006 Rev.~~2~~3. The calculated monitor response is ~~22.4~~ 189 R/hr when RCS level is at the top of the active fuel ~~and 6 R/hr at one foot above the top of active fuel~~. The high range of these monitors is 10 R/hr. The value of 9,000 mR/hr was selected to ensure that the threshold is readily assessable and within the calibrated range of the monitor. The threshold value of 9,000 mR/hr with the reactor head off corresponds to approximately ~~8~~24 inches above the top of the active fuel ~~with the reactor head on~~; which provides an additional indication that RCS levels are near the point of fuel uncovery. These monitor readings in conjunction with the other threshold values allow for an accurate assessment of the EAL.

Core uncovery can be determined by the secondary indications listed in this EAL. The secondary indicators of inventory loss include a list of tanks/sumps found in 0POP04-RC-0003, Excessive RCS Leakage. Since other system leaks could raise levels in various tanks and sumps, the list has been limited to the tanks and sumps that would have the highest probability of indicating RCS leakage inside the Reactor Containment.

REFERENCES:

1. Calculation No: STPNOC013-CALC-006 Rev.~~2~~3, Dose Rate Evaluation of Reactor Vessel Water Levels during Refueling for EAL Thresholds
2. 0POP03-ZG-0009, Rev. 59, Mid-Loop Operation, Addendum 1, RCS/RHR Simplified Elevation Diagram
3. USFAR, Rev. 15, Chapter 12, Table 12.3.4-1
4. 0POP02-II-0002, Rev. 15, RVWL Monitoring System
5. 0POP04-RC-0003, Rev 18, Excessive RCS Leakage
6. Drawing 9C129A81105, Re. 3, Radiation Zones, Reactor Containment Building, Plan at E. 68' – 0"

CG1

ECL: GENERAL EMERGENCY

Initiating Condition: Loss of RCS inventory affecting fuel clad integrity with containment challenged.

Operating Mode Applicability: 5, 6

Emergency Action Levels: (1 or 2)

Note: The Emergency Director should declare the GENERAL EMERGENCY promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.

(1) a. RCS level less than 0% of plenum for 30 minutes or longer.

AND

b. **ANY** indication from the Table C1.

(2) a. RCS level cannot be monitored for 30 minutes or longer.

AND

b. Core uncovery is indicated by **ANY** of the following:

- Reactor Containment Building, 68'-0" Area Radiation Monitors RE-8055 or RE-8099 reading greater than 9,000 mR/hr.

  OR

- Erratic source range monitor indication

  OR

- UNPLANNED rise in **ANY** of the following sump or tank levels in Table C2 of sufficient magnitude to indicate core uncovery

AND

c. **ANY** indication from Table C1

Table C1: Containment Challenge
• CONTAINMENT CLOSURE not established * • ≥4% hydrogen exists inside containment • UNPLANNED rise in containment pressure

* IF CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, THEN declaration of a General Emergency is not required.

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

Basis:

This IC addresses the inability to restore and maintain RCS level above the top of active fuel with containment challenged. This condition represents actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in RCS level. If RCS level cannot be restored, fuel damage is probable.

With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a GENERAL EMERGENCY is not required.

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to Containment integrity.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use indications in Table C1 to assess whether or not containment is challenged.

In EAL 2.b, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS

These EALs address concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.*

CG1: EAL-1 Selection Basis:

Per NEI 99-01 Rev. 6, the RCS level indication should be approximately the top of active fuel (TAF). The RCS level which corresponds to the top of the active fuel is ~~28' 2" (0POP03-ZG-0009, Mid-Loop Operation, Addendum 1, RCS/RHR Simplified Elevation Diagram)~~ 26'-1". The nearest Reactor Vessel Water Level Monitoring System thermocouple to TAF is Sensor 8 at elevation 29'-2.7". Use of RVWL to approximate TAF; with the inherent gap of ~~12~~37 inches between indicated level and actual level, is acceptable for the purposes of maintaining the escalation logic for the loss of RCS level condition.

CG1: EAL-2 Selection Basis:

The secondary indicators of inventory loss include a list of tanks/sumps found in 0POP04-RC-0003, Excessive RCS Leakage. Since other system leaks could rise levels in various tanks and sumps, the list has been limited to the tanks and sumps that would have the highest probability of indicating RCS leakage inside the Reactor Containment Building.

As RCS level drops the dose rates above the core will rise. Area Radiation Monitors RE-8055 and RE-8099 are located on the 68'-0" elevation of the reactor containment building. Their locations are identified on drawing 9C129A81105. Their range (0.1 mR/hr to 10,000 mR/hr) is identified in Table 12.3.4-1 of Section 12 of the UFSAR. Ris~~e~~ing~~s~~ indication on these monitors ~~can be can be an indication that~~ forewarns core uncovery. ~~is occurring.~~ Additionally, erratic source range monitor indications, or large level rises in the tanks listed can give further indication of core uncovery.

The threshold value for radiation monitors RE-8055 and RE-8099 was based on Calculation STPNOC013-CALC-006 Rev. ~~2~~3. The calculated monitor response is ~~22.4~~ 189 R/hr when RCS level is at the top of the active fuel ~~and 6 R/hr at one foot above the top of active fuel~~. The high range of these monitors is 10 R/hr. The value of 9,000 mR/hr was selected for this threshold to ensure the threshold is readily assessable and within the calibrated range of the monitor. The threshold value of 9,000 mR/hr with the reactor head ~~on~~ off corresponds to approximately 24 ~~8~~ inches above the top of the active fuel which provides an additional indication that RCS levels are near the point of fuel uncovery. These monitor readings in conjunction with the other threshold values allow for an accurate assessment of the EAL.

REFERENCES:

1. Calculation No. STPNOC013-CALC-006 Rev. ~~2~~3, Dose Rate Evaluation of Reactor Vessel Water Levels during Refueling for EAL Thresholds
2. 0POP03-ZG-0009, Rev. 59, Mid-Loop Operations
3. Drawing 9C129A81105, Rev. 3, Radiation Zones, Reactor Containment Building Plan at El. 68'-0"
4. USFAR, Rev. 15, Chapter 12, Table 12.3.4-1, Area Radiation Monitors
5. 0POP05-EO-E010, Rev. 21, Loss of Reactor or Secondary Coolant
6. 0POP04-RC-0003, Rev. 18, Excessive RCS Leakage

8 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) ICS/EALS

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

UNUSUAL EVENT

E-HU1 Damage to a loaded cask CONFINEMENT BOUNDARY.
Op. Modes: ALL

E-HU1

ECL: UNUSUAL EVENT

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

Operating Mode Applicability: ALL

Emergency Action Level:

(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
b. 7000 mrem/hr *(gamma + neutron)* on the side surface of the transfer cask.

Basis:

This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage.

The existence of "damage" is determined by radiological survey. The values for this EAL are 2 times the Technical Specification allowable radiation levels. The technical specification multiple of "2 times", which is also used in Recognition Category R IC RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

Security-related events for ISFSIs are covered under ICs HU1 and HA1.

E-HU1 – EAL-1 Selection Basis:

NEI 99-01 Rev.6 states that the dose rate limits are 2 times the Cask Technical Specification Limits. Section 5.3.2 of the "Certificate of Compliance No. 1032, Appendix A, Technical Specifications For The HI-STORM FW MPC Storage System", states:

5.3.4 Notwithstanding the limits established in Section 5.3.3, the measured dose rates on a loaded OVERPACK or TRANSFER CASK shall not exceed the following values:

a. 30 mrem/hr (gamma + neutron) on the top of the OVERPACK

b. 300 mrem/hr (gamma + neutron) on the side of the OVERPACK, excluding inlet and outlet ducts
c. 3500 mrem/hr (gamma + neutron) on the side of the TRANSFER CASK

REFERENCES:

1. Certificate of Compliance no. 1032, Appendix A, Technical Specifications For The HI-STORM FW MPC Storage System, Section 5.3, Radiation Protection Program.10 CFR 72.104, Criteria For Radioactive Materials In Effluents And Direct Radiation From An ISFSI or MRS

9 FISSION PRODUCT BARRIER ICS/EALS

Table 9-F-1: Recognition Category "F" Initiating Condition Matrix

ALERT	
FA1	**ANY** Loss or **ANY** Potential Loss of either the Fuel Clad or RCS barrier. *Op. Modes: 1,2,3,4*
SITE AREA EMERGENCY	
FS1	Loss or Potential Loss of **ANY** two barriers. *Op. Modes: 1,2,3,4*
GENERAL EMERGENCY	
FG1	Loss of **ANY** two barriers and Loss or Potential Loss of the third barrier. *Op. Modes: 1,2,3,4*

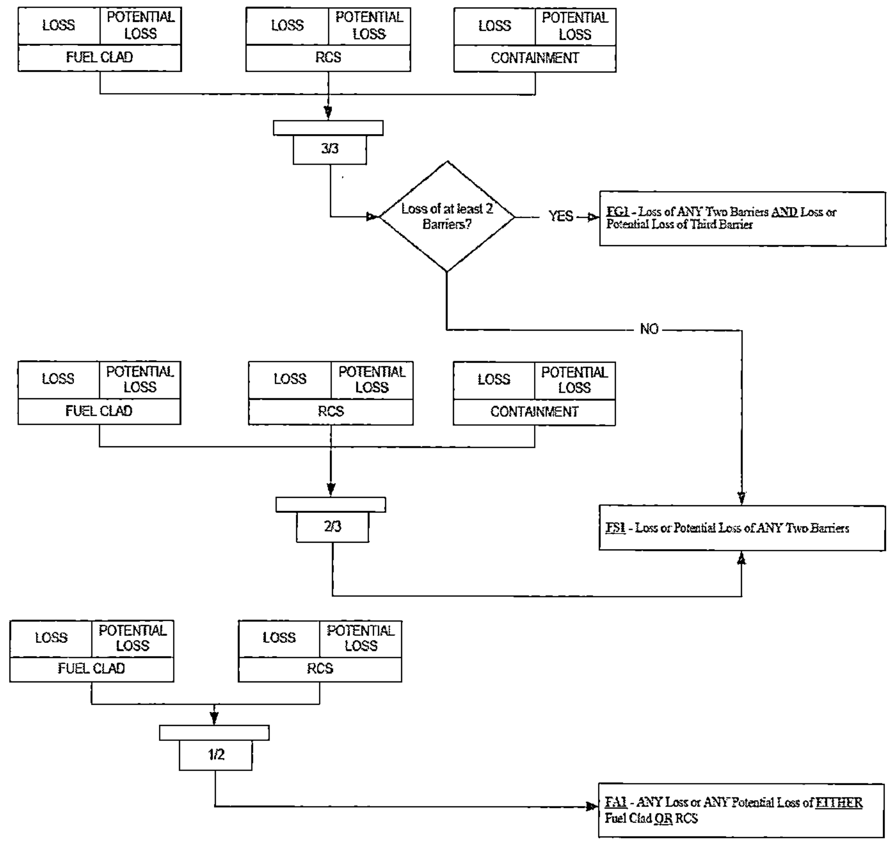


Table 9-F-2: EAL Fission Product Barrier Table

Thresholds for LOSS or POTENTIAL LOSS of Barriers

FA1 ALERT	**FS1 SITE AREA EMERGENCY**	**FG1 GENERAL EMERGENCY**
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.

Fuel Clad Barrier		**RCS Barrier**		**Containment Barrier**	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. RCS or SG Tube Leakage		**1. RCS or SG Tube Leakage**		**1. RCS or SG Tube Leakage**	
Not Applicable	A. Core Cooling - Orange entry conditions met	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	A. A leaking or RUPTURED SG is FAULTED outside of containment.	Not Applicable

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
2. Inadequate Heat Removal		**2. Inadequate Heat Removal**		**2. Inadequate Heat Removal**	
A. Core Cooling - Red entry conditions met	A. Core Cooling - Orange entry conditions met **OR** B. Heat Sink - Red entry conditions met	Not Applicable	A. Heat Sink - Red entry conditions met.	Not Applicable	A. Core Cooling – Red entry conditions met for 15 minutes or longer.
3. RCS Activity / Containment Radiation		**3. RCS Activity / Containment Radiation**		**3. RCS Activity / Containment Radiation**	
A1. RCB Rad Monitor RT-8050 or RT-8051 greater than ~~40~~ 2100 R/hr **OR** 2. HATCH MONITOR greater than ~~90~~ 4200mR/hr **OR** B. Sample analysis indicates that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131.	Not Applicable	A1. RCB Rad Monitor RT-8050 or RT-8051 greater than 10 R/hr **OR** 2. HATCH MONITOR greater than 20mR/hr ~~Not Applicable~~	Not Applicable	Not Applicable	A1. RCB Rad Monitor RT-8050 or RT-8051 greater than ~~380~~45,000 R/hr **OR** 2. HATCH MONITOR greater than ~~840~~ 90,000mR/hr

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
4. Containment Integrity or Bypass		**4. Containment Integrity or Bypass**		**4. Containment Integrity or Bypass**	
Not Applicable	Not Applicable	Not Applicable	Not Applicable	A. Containment isolation is required **AND** EITHER of the following: 1. Containment integrity has been lost based on Emergency Director judgment. **OR** 2. UNISOLABLE pathway from the containment to the environment exists. **OR** B. Indications of RCS leakage outside of containment.	A. Containment - Red entry conditions met **OR** B. Explosive mixture exists inside containment ($H_2 \geq 4\%$) **OR** C1. Containment pressure greater than 9.5 psig. **AND** 2. Less than one full train of Containment Spray is operating per design for 15 minutes or longer.

Fuel Clad Barrier		RCS Barrier		Containment Barrier	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
5. Other Indications		**5. Other Indications**		**5. Other Indications**	
A. N/A	A. N/A	A. N/A	A. N/A	A. N/A	A. N/A
6. Emergency Director Judgment		**6. Emergency Director Judgment**		**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 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 Containment Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Basis Information For
EAL Fission Product Barrier Table 9-F-2

STP 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.

FUEL CLAD BARRIER THRESHOLDS

The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.

1. RCS or SG Tube Leakage

Loss 1

There is no Loss threshold associated with RCS or SG Tube Leakage.

Potential Loss 1.A

Core Cooling - Orange entry conditions (CETs ≥ 708° F) are sufficient to allow the onset of heat-induced cladding damage.

2. Inadequate Heat Removal

Loss 2.A

Core Cooling - Red entry conditions (CETs ≥ 1200° F) are sufficient to cause significant superheating of reactor coolant.

Potential Loss 2.A

Core Cooling - Orange entry conditions (CETs ≥ 708° F) are sufficient to allow the onset of heat-induced cladding damage.

Potential Loss 2.B

Heat Sink - Red entry conditions met (NR level in all SG ≤ 14% [34%] AND total AFW flow to SG ≤ 576 GPM). This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the Fuel Clad Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

Meeting this threshold results in a SITE AREA EMERGENCY because this threshold is identical to RCS Barrier Potential Loss threshold 2.A; both will be met. This condition warrants a SITE AREA EMERGENCY declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and raise RCS pressure to the point where mass will be lost from the system.

FUEL CLAD BARRIER THRESHOLDS

3. RCS Activity / Containment Radiation

Loss 3.A~~-~~1

The readings for the containment high range area monitors (RT-8050 and RT-8051) correspond to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier. The values for RT-8050 and RT-8051 were based on Calculation ~~STPNOC013-004 Rev.2~~15-RA-011. The threshold values were conservatively rounded down from the calculated value of 2144 R/hr~~within 2% of the calculated values~~ to make the values readily assessable. Temperature induced current (TIC) limitations are not applicable to the Fuel Clad Barrier Loss threshold 3.A.1 because the expected radiation dose for this event overwhelms the TIC effect. This effect is discussed in the 10CFR50.59 evaluation 04-8245-60 associated with DCP 04-8245-33.

Loss 3.A~~-~~2

The HATCH MONITOR is located outside containment and is the back-up monitor to the containment high range monitors (RT-8050 and RT-8051). The HATCH MONITOR threshold value is based on Calculation No. 03-ZE-003. This value corresponds to the calculated containment high range monitor readings for Fuel Clad Barrier Loss 3.A

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold 3.A since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the EMERGENCY CLASSIFICATION LEVEL to a SITE AREA EMERGENCY.

Loss 3.B

This threshold indicates that RCS radioactivity concentration is greater than 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

Potential Loss 3.

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

4. Containment Integrity or Bypass

Not Applicable (included for numbering consistency)

FUEL CLAD BARRIER THRESHOLDS

5. Other Indications

Loss and/or Potential Loss 5.A

N/A

6. Emergency Director Judgment

Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad Barrier is lost.

Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

RCS BARRIER THRESHOLDS

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

1. RCS or SG Tube Leakage

Loss 1.A

This threshold is based on an UNISOLABLE RCS leak of sufficient size to require an automatic or manual actuation of the Emergency Core Cooling System (ECCS). This condition clearly represents a loss of the RCS Barrier.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED. If a RUPTURED steam generator is also FAULTED outside of containment, the declaration escalates to a SITE AREA EMERGENCY since the Containment Barrier Loss threshold 1.A will also be met.

Potential Loss 1.A

This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ECCS (SI) actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs that a standby charging (makeup) pump be placed in service to restore and maintain pressurizer level.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

If a leaking steam generator is also FAULTED outside of containment, the declaration escalates to a SITE AREA EMERGENCY since the Containment Barrier Loss threshold 1.A will also be met.

Potential Loss 1.B

Integrity – Red entry conditions indicate an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized).

RCS BARRIER THRESHOLDS

2. Inadequate Heat Removal

Loss 2.A

There is no Loss threshold associated with Inadequate Heat Removal.

Potential Loss 2.A

Heat Sink – Red entry conditions met (NR level in all SG ≤ 14% [34%] AND total AFW flow to SGs ≤ 576 GPM).

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the RCS Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

Meeting this threshold results in a SITE AREA EMERGENCY because this threshold is identical to Fuel Clad Barrier Potential Loss threshold 2.B; both will be met. This condition warrants a SITE AREA EMERGENCY declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and raise RCS pressure to the point where mass will be lost from the system.

3. RCS Activity / Containment Radiation

Loss 3.A1.

Calculation 15-RA-11 provides a value that corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals the Technical Specification allowable limits. The threshold values were conservatively rounded down from the calculated value of 13 R/hr to make the values readily assessable. Temperature induced current (TIC) limitations are not applicable to the RCS Barrier Loss threshold 3.A1 because the expected radiation dose for this event overwhelms the TIC effect. This effect is discussed in the 10CFR50.59 evaluation 04-8245-60 associated with DCP 04-8245-33.

~~Not Applicable~~

Loss 3.A2

The HATCH MONITOR is located outside containment and is the back-up monitor to the containment high range monitors (RT-8050 and RT-8051). The HATCH MONITOR threshold value is based on Calculation No. 03-ZE-003. This value corresponds to the calculated containment high range monitor readings for RCS Barrier Loss 3.A1

Potential Loss 3.

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

RCS BARRIER THRESHOLDS

4. Containment Integrity or Bypass

Not Applicable (included for numbering consistency)

5. Other Indications

Loss and/or Potential Loss 5.A

Variables used to monitor for the significant breach or the potential significant breach of fuel clad, the RCS pressure boundary, or the reactor Containment, are designated Type C. The response characteristics of Type C information display channels allow the control room operator to detect conditions indicative of significant failure of any of the three fission product barriers or the potential for significant failure of these barriers. Although variables selected to fulfill Type C functions may rapidly approach the values that indicate an actual significant failure, it is the final steady-state value reached that is important. Therefore, a high degree of accuracy and a rapid response time are not necessary for Type C information display channels. Type C variables are found in UFSAR Table 7B.6-1.

6. Emergency Director Judgment

Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is lost.

Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

CONTAINMENT BARRIER THRESHOLDS

The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the ECL from ALERT to a SITE AREA EMERGENCY or a GENERAL EMERGENCY.

1. RCS or SG Tube Leakage

Loss 1.A

This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for RCS Barrier Potential Loss 1.A and Loss 1.A, respectively. This condition represents a bypass of the containment barrier.

FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably [*part of the FAULTED definition*] and the FAULTED steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition, the steam generator is still considered FAULTED for emergency classification purposes.

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU3 for the fuel clad barrier (i.e., RCS activity values) and IC SU4 for the RCS barrier (i.e., RCS leak rate values).

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, glad seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

CONTAINMENT BARRIER THRESHOLDS

The EMERGENCY CLASSIFICATION LEVELS resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

Affected SG is FAULTED Outside of Containment?

P-to-S Leak Rate	**Yes**	**No**
Less than or equal to 25 gpm	No classification	No classification
Greater than 25 gpm	UNUSUAL EVENT per SU4	UNUSUAL EVENT per SU4
Requires operation of a standby charging pump (*RCS Barrier Potential Loss*)	SITE AREA EMERGENCY per FS1	ALERT per FA1
Requires an automatic or manual ECCS (SI) actuation (*RCS Barrier Loss*)	SITE AREA EMERGENCY per FS1	ALERT per FA1

Potential Loss 1.

There is no Potential Loss threshold associated with RCS or SG Tube Leakage.

2. Inadequate Heat Removal

Loss 2

There is no Loss threshold associated with Inadequate Heat Removal.

Potential Loss 2.A

Core Cooling – Red entry conditions met for 15 minutes or longer. This condition represents an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and a higher potential for containment failure. For this condition to occur there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.

The restoration procedure is considered "effective" if core exit thermocouple readings are decreasing and/or if RCS level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The Emergency Director should escalate the emergency classification level as soon as it is determined that the procedure(s) will not be effective.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation in a significant fraction of core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.

CONTAINMENT BARRIER THRESHOLDS

3. RCS Activity / Containment Radiation

Loss 3

There is no Loss threshold associated with RCS Activity / Containment Radiation.

Potential Loss 3.A.1

The readings for the containment high range area monitors (RT-8050 and RT-8051) correspond to an instantaneous release of the radioactive material inventory of the reactor coolant system (i.e., All the RCS coolant mass) into the containment, assuming that 20% of the fuel cladding has failed. The values for RT-8050 and RT-8051 were based on Calculation No.~~ STPNOC013-004 Rev.2~~15-RA-11. The threshold values used were conservatively rounded ~~within 2% of~~ down from the calculated value~~s~~ of 45,040 R/hr to ensure the values were readily assessable. This level of assumed fuel clad failure is well beyond that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds. Temperature induced current (TIC) limitations are not applicable to the Containment Barrier Potential Loss threshold 3.A.1 because the expected radiation dose for this event overwhelms the TIC effect. This effect is discussed in 10CFR50.59 evaluation 04-8245-60 associated with DCP 04-8245-33.

NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the EMERGENCY CLASSIFICATION LEVEL to a GENERAL EMERGENCY.

Potential Loss 3.A.2

The HATCH MONITOR is located outside containment and is the back-up monitor to the containment high range monitors (RT-8050 and RT-8051). The HATCH MONITOR threshold value is based on Calculation No. 03-ZE-003. This value corresponds to the calculated containment high range monitor readings for Containment Barrier Threshold Potential Loss 3.A.1.

4. Containment Integrity or Bypass

Loss 4.A

These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both thresholds 4.A.1 and 4.A.2.

4.A.1 – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate

during accident conditions, it is expected that the Emergency Director will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure 9-F-3. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

Another example would be a loss or potential loss of the RCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

4.A.2 – Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment atmosphere to the environment. As used here, the term "environment" includes the atmosphere of a room or area, outside the containment, that may, in turn, communicate with the outside-the-plant atmosphere (e.g., through discharge of a ventilation system or atmospheric leakage). Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.

Refer to the top piping run of Figure 9-F-3 in Addendum 3, Containment Integrity or Bypass Examples. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Leakage between two interfacing liquid systems, by itself, does not meet this threshold.

Refer to the bottom piping run of Figure 9-F-3. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the Component Cooling Water system to the Auxiliary Building, then no threshold has been met. If the pump or system piping developed a leak that allowed steam/water to enter the Auxiliary Building, then threshold 4.B would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause threshold 4.A.1 to be met as well.

CONTAINMENT BARRIER THRESHOLDS

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. Minor releases may also occur if a containment isolation valve(s) fails to close but the containment atmosphere escapes to a closed system. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold 1.A.

Loss 4.B

Containment sump, temperature, pressure and/or radiation levels will rise if reactor coolant mass is leaking into the containment. If these parameters have not risen, then the reactor coolant mass may be leaking outside of containment (i.e., a containment bypass sequence). Rises in sump, temperature, pressure, flow and/or radiation level readings outside of the containment may indicate that the RCS mass is being lost outside of containment.

Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of RCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside of containment may not rise significantly; however, other unexpected changes in sump levels, area temperatures or pressures, flow rates, etc. should be sufficient to determine if RCS mass is being lost outside of the containment.

Refer to the middle piping run of Figure 9-F-3. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause threshold 4.A.1 to be met as well.

To ensure proper escalation of the emergency classification, the RCS leakage outside of containment must be related to the mass loss that is causing the RCS Loss and/or Potential Loss threshold 1.A to be met.

Potential Loss 4.A

Containment – Red entry conditions met (containment pressure ≥ 56.5 PSIG). If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the RCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a SITE AREA EMERGENCY and GENERAL EMERGENCY since there is now a potential to lose the third barrier.

Potential Loss 4.B

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit (4%)). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a potential loss of the Containment Barrier.

CONTAINMENT BARRIER THRESHOLDS

Potential Loss 4.C

This threshold describes a condition where containment pressure is greater than the setpoint (9.5 PSIG) at which Containment Spray is 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. This threshold represents a potential loss of containment in that Containment Spray is either lost or performing in a degraded manner.

5. Other Indications

Loss and/or Potential Loss 5.A
N/A

6. Emergency Director Judgment

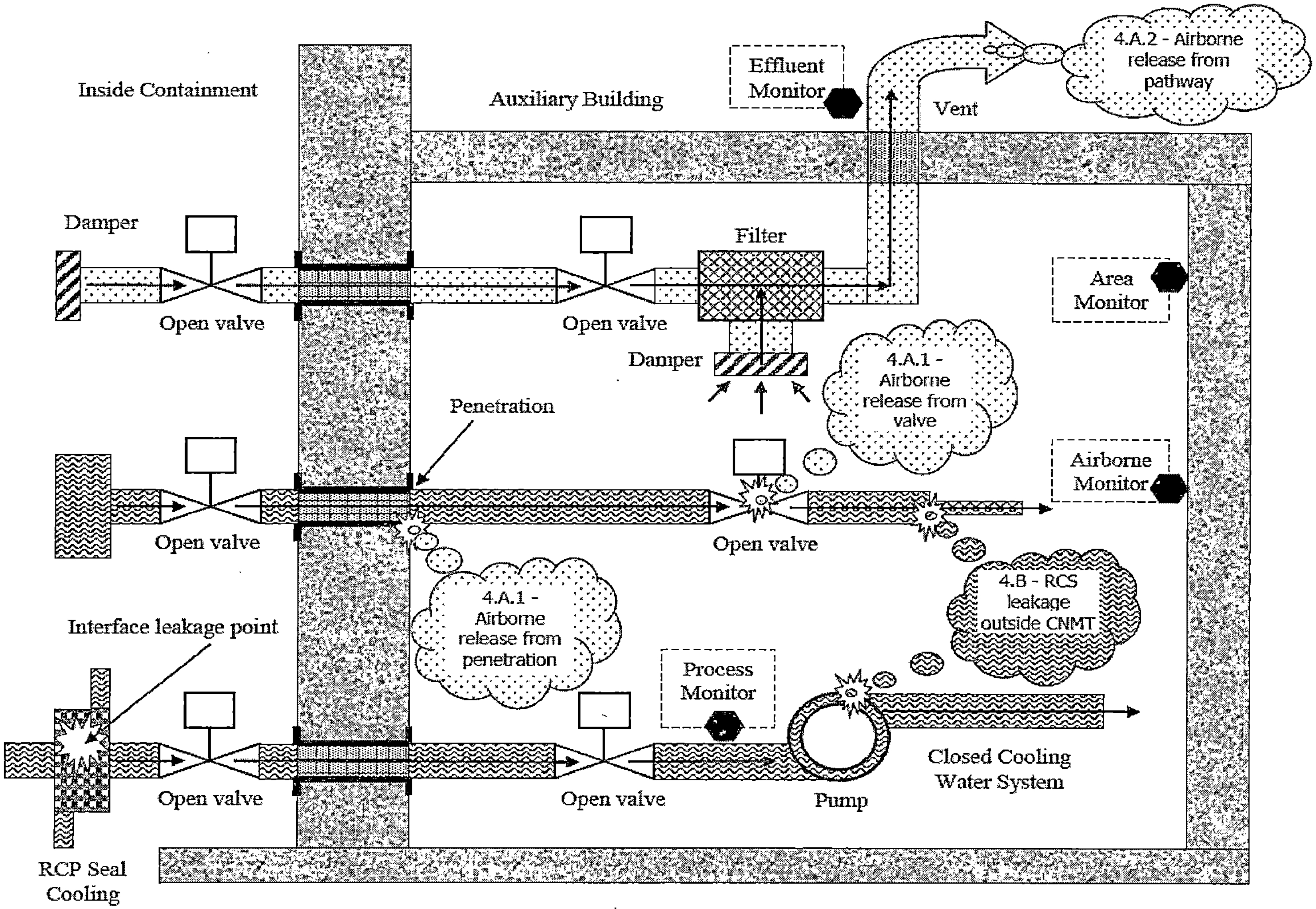
Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is lost.

Potential Loss 6.A

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Figure 9-F-3: Containment Integrity or Bypass Examples



NOTES: Only Supplemental Purge is a filtered release and STPEGS Component Cooling Water is equivalent to Closed Cooling Water

10 HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY ICS/EALS

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

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
HU1 Confirmed SECURITY CONDITION or threat. *Op. Modes: **ALL***	**HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes. *Op. Modes: **ALL***	**HS1** HOSTILE ACTION within the PROTECTED AREA. *Op. Modes: **ALL***	**HG1** HOSTILE ACTION resulting in loss of physical control of the facility. *Op. Modes: **ALL***
HU2 Seismic event greater than OBE levels. *Op. Modes: **ALL*** **HU3** Hazardous event. *Op. Modes: **ALL*** **HU4** FIRE potentially degrading the level of safety of the plant. *Op. Modes: **ALL***	**Note:** **See SA9 or CA6 for escalation of these events**		
	HA5 Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown. *Op. Modes: **ALL***		
	HA6 Control Room evacuation resulting in transfer of plant control to alternate locations. *Op. Modes: **ALL***	**HS6** Inability to control a key safety function from outside the Control Room. *Op. Modes: **ALL***	
HU7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of an UNUSUAL EVENT. *Op. Modes: **ALL***	**HA7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of an ALERT. *Op. Modes: **ALL***	**HS7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY. *Op. Modes: **ALL***	**HG7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY. *Op. Modes: **ALL***

ECL: UNUSUAL EVENT

Initiating Condition: Confirmed SECURITY CONDITION or threat.

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2 or 3)

(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

(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 SAFETY SYSTEM equipment, 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 ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Force 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 Security Force Supervisor 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 § 2.39039 information.

EAL #2- addresses the receipt of a CREDIBLE SECURITY THREAT. The credibility of the threat is assessed in accordance with 0SDP01-ZS-0011, Implementing Procedure For Safeguards Contingency Events.

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 0POP04-ZO-SEC4, Guideline For Airborne (Aircraft) Threat, and Security Force Instruction SI 2700, Security Response to Airborne Threat.

Emergency plans and implementing procedures are public documents; therefore, EALs 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 is contained in the Security Plan.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC HA1.

HU1: EAL-1 Selection Basis:

For EAL-1, the position of Security Force Supervisor was included since it is a 24-hour position. Normally the event would not be reported by the Acting Security Manager or Security Manager because the Acting Security Manager position is not normally activated until after an UNUSUAL EVENT has been declared, and the Security Manager position is not normally activated until after an ALERT has been declared. However, reporting by the Acting Security Manager or Security Manager was included in the event these positions are staffed under unusual circumstances.

REFERENCEs:

1. 0ERP01-ZV-SH03, Rev. 12, Acting Security Manager
2. 0ERP01-ZV-TS08, Rev. 16, Security Manager
3. 0POP04-ZO-SEC4, Rev. 10, Guideline For Airborne (Aircraft) Threat (SUNSI)
4. 0SDP01-ZS-0011, Implementing Procedure For Safeguards Contingency Events (Safeguards)
5. Security Force Instruction SI 2700, Security Response to Airborne Threat (SUNSI)

ECL: UNUSUAL EVENT

Initiating Condition: Seismic event greater than OBE levels.

Operating Mode Applicability: ALL

Emergency Action Level:

(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 manner deemed appropriate by the Shift Manager or Emergency Director.

Basis:

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE). An earthquake greater than an OBE but less than a Safe Shutdown Earthquake (SSE) should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., performs walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Although the "SEISMIC EVENT" alarm (0.02 g) in EAL 1.a is set below an O.B.E earthquake (0.05 g), it does provide an indication that a seismic event has occurred. In order to determine whether an O.B.E. earthquake occurred, additional indications may be needed. Determination per 0POP04-SY-001, Seismic Event is not practical if it takes longer than 15 minutes to perform.

Indications described in the EAL should be limited to those that are immediately available to Control Room personnel and which can be readily assessed. Indications available outside the Control Room and/or which require lengthy times to assess (e.g., processing of scratch plates or recorded data) should not be used. The goal is to specify indications that can be assessed within 15-minutes of the actual or suspected seismic event.

The EAL 1.b- statement is included to ensure that a declaration does not result from felt vibrations caused by a non-seismic source (e.g., a dropped heavy load). The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration. It is recognized that this alternate EAL wording may cause a site to declare an UNUSUAL EVENT while another site, similarly affected but with readily assessable OBE indications in the Control Room, may not.

Depending upon the plant mode at the time of the event, escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC CA6 or SA9.

HU2: EAL-1 Selection Basis:

STP does not have a readily available indication in the Control Room for determining if the site has experienced an OBE. The 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 STP has experienced an OBE.

STP has implemented the alternative EAL described in NEI 99-01 Developer Notes in conjunction with using the installed indication. EAL-1, b. allows the Shift Manager or Emergency Director to determine if a seismic event has taken place, taking into consideration the Seismic Event alarm, Control Room personnel feeling an actual or potential seismic event and other indications deemed appropriate.

REFERENCES:

1. 0POP04-SY-0001, Rev. 8, Seismic Event
2. NEI 99-01, Rev. 6, Development of Emergency Action Levels for Non-Passive Reactors.

HU3

ECL: UNUSUAL EVENT

Initiating Condition: Hazardous event.

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2 or 3 or 4 or 5)

Note: EAL #4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

(1) A tornado strike within the PROTECTED AREA.

(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.

(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).

(4) A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.

(5) Predicted or actual breach of Main Cooling Reservoir retaining dike along North Wall

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

EAL #1- addresses a tornado striking (touching down) within the PROTECTED AREA.

EAL #2- addresses flooding of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

EAL #3- addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.

EAL #4- addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site flooding caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road. This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around

the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.

EAL#5- the Main Cooling Reservoir breach along the north wall which was included because it is a credible hazard and analyzed in the STPEGS UFSAR.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be based on ICs in Recognition Categories R, F, S or C.

HU3: EAL-1, EAL-2, EAL-3, EAL-4 Selection Basis:

N/A

REFERENCE:

1. STPEGS UFSAR, Section 3.4.1, Flood Protection

HU4

ECL: UNUSUAL EVENT

Initiating Condition: FIRE potentially degrading the level of safety of the plant.

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2 or 3 or 4)

Note: The Emergency Director should declare the UNUSUAL EVENT promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

(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

b. The FIRE is located within **ANY** of the plant rooms or areas in Table H4:

Table H4: 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)

(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 H4:

AND

c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.

(3) A FIRE within the ISFSI **OR** plant PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.

(4) A FIRE within the ISFSI **OR** plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

EAL #1

The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

EAL #2

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then EAL #1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

EAL #3

In addition to a FIRE addressed by EAL #1 or EAL #2, a FIRE within the plant or ISFSI PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety.

EAL #4

If a FIRE within the plant or ISFSI PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.

Basis-Related Requirements from Appendix R

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and EXPLOSIONS."
>
> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.
>
> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

Depending upon the plant mode at the time of the event, escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC CA6 or SA9.

HU4: EAL-1.b, EAL-2.b Selection Basis:

The plant areas or rooms listed contain SAFETY SYSTEM equipment.

REFERENCES:

1. 0PGP03-ZF-0001, Rev. 26, Fire Protection Program
2. STPEGS UFSAR, Rev. 16, Section 7.4, Systems Required for Safe Shutdown

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: ALL

Emergency Action Level:

(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 an UE.

HU7: EAL-1 Selection Basis:

N/A

REFERENCES:

N/A

HA1

ECL: ALERT

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

Operating Mode Applicability: ALL

Emergency Action Levels: (1 or 2)

(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

(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.

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 0POP04-ZO-SEC4, Guidelines for Airborne (Aircraft) Threat, and Security Force Instruction SI 2700, Security Response to Airborne Threat.

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 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 is contained in the Security Plan.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via IC HS1.

HA1: EAL-1 and EAL-2 Selection Basis:

The EALs are taken from NEI 99-01, Rev. 6. For EAL-1, the positions of Security Force Supervisor OR Acting Security Manager were included because either of these positions could be activated prior to meeting this EAL. The Security Force Supervisor is a 24-hour position and the normally the Acting Security Manager is activated after an UNUSUAL EVENT has been declared. The Security Manager is also included although this position is normally activated after an ALERT.

REFERENCES:

1. 0ERP01-ZV-SH03, Rev. 12, Acting Security Manager
2. 0ERP01-ZV-TS08, Rev. 16, Security Manager
3. 0POP04-ZO-SEC4, Rev. 10, Guideline For Airborne (Aircraft) Threat (SUNSI)
4. Security Force Instruction SI 2700, Security Response to Airborne Threat (SUNSI)

HA5

ECL: ALERT

Initiating Condition: Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown.

Operating Mode Applicability: ALL

Emergency Action Level:

Note: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

(1) a. Release of a toxic, corrosive, asphyxiant or flammable gas into the Control Room or **ANY** of the plant rooms or areas listed in Table H3/R2:

AND

b. Entry into the room or area is prohibited or impeded.

TABLE H3/R2: Plant Areas Requiring Access		
MODE 4	RCB	RHR Heat Exchanger Rooms
	MAB	51 ft Room 335
	EAB	Roof, MCC 1G8, 4.16KV Switchgear Rooms
MODE 5	EAB	4.16KV Switchgear Rooms

Basis:

This IC addresses an event involving a release of a hazardous gas that precludes or impedes access to equipment necessary to maintain normal plant operation, or required for a normal plant cooldown and shutdown. This condition represents an actual or potential substantial degradation of the level of safety of the plant.

An ALERT declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.

Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the Emergency Director's judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

An emergency declaration is not warranted if any of the following conditions apply.

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release).
- For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

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

This EAL does not apply to firefighting activities that automatically or manually activate a fire suppression system in an area.

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via Recognition Category R, C or F ICs.

HA5: EAL-1 Selection Basis:

The areas listed in EAL-1 apply to areas that contain equipment necessary for plant operations, cooldown, or shutdown. Assuming all plant equipment is operating as designed, Normal operations and safe shutdown equipment operation is capable from the Main Control Room (MCR). The plant is able to transition into a hot shutdown from the MCR, therefore H3/R2 is a list of plant rooms or areas with entry-related mode applicability that contain equipment which require a manual/local action necessary following entry into hot shutdown (establish Residual Heat Removal shutdown cooling, disable operation of charging and ECCS equipment, and limit dilution pathways) and subsequent entry into cold shutdown (disable operation of ECCS equipment). After achieving cold shutdown it is assumed that the plant will be maintained in a cold shutdown condition.

REFERENCES:

1. 0PGP03-ZF-0001, Rev. 26, Fire Protection Program
2. STPEGS UFSAR, Rev. 16, Section 7.4, Systems Required for Safe Shutdown
3. 0POP03-ZG-0008, Rev. 56, Power Operations
4. 0POP03-ZG-0006, Rev. 54, Plant Shutdown from 100% to Hot Standby
5. 0POP03-ZG-0007, Rev. 71, Plant Cooldown

HA6

ECL: ALERT

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

Operating Mode Applicability: ALL

Emergency Action Level:

(1) An event has resulted in plant control being transferred from the Control Room to 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 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

ECL: ALERT

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

Operating Mode Applicability: ALL

Emergency Action Level:

(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.

HA7: EAL-1 Selection Basis:

N/A

REFERENCE:

N/A

ECL: SITE AREA EMERGENCY

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA.

Operating Mode Applicability: ALL

Emergency Action Level:

(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

<u>Basis:</u>

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 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 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 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 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 is contained in the Security Plan.

Escalation of the 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

ECL: SITE AREA EMERGENCY

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

Operating Mode Applicability: ALL

Emergency Action Level:

Note: The Emergency Director should declare the SITE AREA EMERGENCY promptly upon determining that 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 the Auxiliary Shutdown Panel (ASP).

AND

b. Control of **ANY** of the following key safety functions in Table H2 is not reestablished within 15 minutes in Modes 1, 2 or 3 ONLY.

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 Auxiliary Shutdown Panel is based on Emergency Director judgment. The Emergency Director is expected to make a reasonable, informed judgment within 15 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 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. The mode applicability conditioning statement for Table H2 is based on the Technical Specification Operability requirement for the following functions of the Remote Shutdown System:

- Core reactivity control (initial and long term)

- RCS pressure control
- Decay heat removal via the AFW System and the SG safety valves or SG PORVs
- RCS inventory control via charging flow, and
- Safety support systems for the above functions.

REFERENCE:

1. Procedure 0POP04-ZO-0001, Rev. 35, Control Room Evacuation
2. Technical Specification 3.3.3.5 Remote Shutdown System

ECL: SITE AREA EMERGENCY

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

Operating Mode Applicability: ALL

Emergency Action Level:

(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.

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 SITE AREA EMERGENCY.

HS7: EAL-1 Selection Basis:

N/A

REFERENCE:

N/A

HG1

ECL: GENERAL EMERGENCY

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

Operating Mode Applicability: ALL

Emergency Action Level:

(1) a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by **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 in MODES 1, 2 or 3 ONLY.

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

OR

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

<u>Basis:</u>

This IC addresses an event in which a HOSTILE FORCE has taken physical control of the 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 Security Program]*.

Emergency plans and implementing procedures are public documents; therefore, EALs 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 is contained in 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. The mode applicability conditioning statement for Table H2 is based on the Technical Specification Operability requirement for the following Functions of the Remote Shutdown System:

- Core reactivity control (initial and long term)
- RCS pressure control
- Decay heat removal via the AFW System and the SG safety valves or SG PORVs
- RCS inventory control via charging flow, and
- Safety support systems for the above Functions.

REFERENCES:

1. 0ERP01-ZV-SH03, Rev. 12, Acting Security Manager
2. 0ERP01-ZV-TS08, Rev. 16, Security Manager
3. Technical Specification 3.3.3.5 Remote Shutdown System

ECL: GENERAL EMERGENCY

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

Operating Mode Applicability: ALL

Emergency Action Level:

(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.

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 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** offsite AC power capability to emergency buses for 15 minutes or longer. *Op. Modes: 1,2,3,4*	**SA1** Loss of **ALL** but one AC power source to emergency buses for 15 minutes or longer. *Op. Modes: 1,2,3,4*	**SS1** Loss of **ALL** offsite and **ALL** onsite AC power to emergency buses for 15 minutes or longer. *Op. Modes: 1,2,3,4*	**SG1** Prolonged loss of **ALL** offsite and **ALL** onsite AC power to emergency buses. *Op. Modes: 1,2,3,4*
SU2 UNPLANNED loss of Control Room indications for 15 minutes or longer. *Op. Modes: 1,2,3,4*	**SA2** UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress. *Op. Modes: 1,2,3,4*		
SU3 Reactor coolant activity greater than Technical Specification allowable limits. *Op. Modes: 1,2,3,4*			
SU4 RCS leakage for 15 minutes or longer. *Op. Modes: 1,2,3,4*			
SU5 Automatic or manual trip fails to shutdown the reactor. *Op. Modes: 1,2*	**SA5** 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. *Op. Modes: 1,2*	**SS5** Inability to shutdown the reactor causing a challenge to core cooling or RCS heat removal. *Op. Modes: 1,2*	

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

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

SU1

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

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

(1) Loss of **ALL** offsite AC power capability to **ALL** three 4160V AC ESF Buses 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 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

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

Note: The Emergency Director should declare the 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.

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 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 RCS level 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 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

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels: (1 or 2)

(1) RT-8039 reading greater than 30 $\mu Ci/cm^3$.

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

- Greater than 1μCi/gm Dose Equivalent I-131
- Greater than 100/ Ē 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 would be via ICs FA1 or the Recognition Category 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 $\mu Ci/cm^3$ 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 Rev.1, Gross Failed Fuel Monitor Response to Rise RCS Activity (RT-8039 EAL Threshold)
2. STP Technical Specification Section 3/4.4.8 Specific Activity.

SU4

ECL: UNUSUAL EVENT

Initiating Condition: RCS leakage for 15 minutes or longer.

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels: (1 or 2 or 3)

Note: The Emergency Director should declare the 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 10 gpm for 15 minutes or longer.

(2) RCS identified leakage greater than 25 gpm for 15 minutes or longer.

(3) Leakage from the RCS to a location outside containment greater than 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) 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. An 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).

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

Escalation of the EMERGENCY CLASSIFICATION LEVEL would be via ICs of Recognition Category 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 Specification limit for primary-to-secondary leakage is 150 gallons per day through any one steam generator, but the specification does not specify the type of leakage. Therefore, STPEGS will use the leakage outside containment; which may include SG Tube Leakage, at 25 gpm for 15 minutes or longer in accordance with NEI 99-01 Rev. 6 guidance.

EFERENCES:

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

ECL: UNUSUAL EVENT

Initiating Condition: Automatic or manual trip fails to shutdown the reactor.

Operating Mode Applicability: 1, 2

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 did not shutdown the reactor.

AND

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

(2) a. A manual trip did not shutdown the reactor.

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

2. A subsequent automatic trip 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 that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control panels or an automatic trip 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, operators will promptly initiate manual actions at the reactor control panels to shutdown the reactor (e.g., initiate a manual reactor trip). 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 is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control panels to shut down the reactor (e.g., initiate a manual reactor trip) using a different switch). Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a

subsequent manual or automatic trip 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 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). 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 panels".

The plant response to the failure of an automatic or manual reactor trip 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 panels are also unsuccessful in shutting down the reactor, then the EMERGENCY CLASSIFICATION LEVEL will escalate to an 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 declaration is appropriate for this event.

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

Should a reactor trip 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 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 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%

ECL: UNUSUAL EVENT

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels: (1 or 2 or 3)

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

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

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

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 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

ECL: UNUSUAL EVENT

Initiating Condition: Failure to isolate containment or loss of containment pressure control.

Operating Mode Applicability: 1, 2, 3, 4

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 within 15 minutes of the actuation signal.

(2) a. Containment pressure greater than 9.5 psig.

AND

b. No 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 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.

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) are either lost or performing in a degraded manner.

This event would escalate to a 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

ECL: ALERT

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

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

(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 a loss of **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 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 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 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

ECL: ALERT

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

Note: The Emergency Director should declare the 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.

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 trip
- ECCS (SI) actuation

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 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 RCS level 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 would be via ICs FS1 or IC 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

SA5

ECL: ALERT

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.

Operating Mode Applicability: 1, 2

Emergency Action Level:

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 did not shutdown the reactor.

AND

b. Manual actions taken at the reactor control 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 that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control 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 panels since this event entails a significant failure of the RPS.

A manual action at the reactor control 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). 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 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 panels".

The plant response to the failure of an automatic or manual reactor trip 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 or RCS heat removal safety functions, the EMERGENCY CLASSIFICATION LEVEL will escalate to a SITE AREA EMERGENCY via IC SS5. Depending upon plant responses and symptoms, escalation is also possible via IC FS1.

It is recognized that plant responses or symptoms may also require an 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

ECL: ALERT

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

(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

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 would be via IC FS1 or RS1.

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

ECL: SITE AREA EMERGENCY

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

Note: The Emergency Director should declare the 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 **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 would be via ICs RG1, FG1 or SG1.

SS1: EAL-1 Selection Criteria:

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

ECL: SITE AREA EMERGENCY

Initiating Condition: Inability to shutdown the reactor causing a challenge to core cooling or RCS heat removal.

Operating Mode Applicability: 1, 2

Emergency Action Level:

(1) a. An automatic or manual trip did not shutdown the reactor.

AND

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

AND

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

- Core Cooling – Red entry conditions met

OR

- Heat Sink- Red entry conditions met

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip 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.

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 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 would be via IC RG1RG1 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.

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

SS8

ECL: SITE AREA EMERGENCY

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

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

(1) Indicated voltage is less than 105.5 VDC on **ALL** 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 would be via ICs 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

SG1

ECL: GENERAL EMERGENCY

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Level:

Note: The Emergency Director should declare the GENERAL EMERGENCY promptly upon determining that 4 hours has been exceeded, or will likely be exceeded.

(1) a. Loss of **ALL** offsite and **ALL** onsite AC power to **ALL** three 4160V AC ESF Buses.

AND

b. **EITHER** of the following:

- Restoration of at least one 4160VAC ESF bus in less than 4 hours is not likely.
- 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 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 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. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is a 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 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

ECL: GENERAL EMERGENCY

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

Operating Mode Applicability: 1, 2, 3, 4

Emergency Action Levels:

Note: The Emergency Director should declare the 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 **ALL** three 4160V AC ESF buses for 15 minutes or longer.

AND

b. Indicated voltage is less than 105.5 VDC on **ALL** 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

APPENDIX A – ACRONYMS AND ABBREVIATIONS

AC	Alternating Current
AOP	Abnormal Operating Procedure
ATWS	Anticipated Transient Without Scram
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
ERG	Emergency Response Guideline
FEMA	Federal Emergency Management Agency
FSAR	Final Safety Analysis Report
GE	GENERAL EMERGENCY
IC	Initiating Condition
ID	Inside Diameter
ISFSI	Independent Spent Fuel Storage Installation
Keff	Effective Neutron Multiplication Factor
LCO	Limiting Condition of Operation
LOCA	Loss of Coolant Accident
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	Nuclear Management and Resources Council
OBE	Operating Basis Earthquake
OCA	Owner Controlled Area
ODCM	Offsite Dose Calculation Manual
ORO	Off-site Response Organization
PA	Protected Area
PAG	Protective Action Guideline
PRA/PSA	Probabilistic Risk Assessment / Probabilistic Safety Assessment

PWR....................Pressurized Water Reactor
PSIG....................Pounds per Square Inch Gauge
R....................Roentgen
RCS....................Reactor Coolant System
Rem, rem, REM....................Roentgen Equivalent Man
RPS....................Reactor Protection System
RPV....................Reactor Pressure Vessel
RVWL....................Reactor Vessel Water Level
SAR....................Safety Analysis Report
SCBA....................Self-Contained Breathing Apparatus
SG....................Steam Generator
SI....................Safety Injection
SPDS....................Safety Parameter Display System
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: 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: 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.

UNUSUAL EVENT UE: 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: 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 emergency classification scheme.

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 (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)

FISSION PRODUCT BARRIER THRESHOLD: A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.

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.

Selected terms used in INITIATING CONDITION and 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: The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.

CONTAINMENT CLOSURE: Those actions necessary to place the RCB in the closed containment condition that provides at least one integral barrier to the release of radioactive material. Sufficient separation of the containment atmosphere from the outside environment is to be provided such that a barrier to the escape of radioactive material is reasonably expected to remain in place following a core melt accident.

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

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: 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: The area under continuous access monitoring and control, and armed protection as described in the site Security Plan.

REFUELING PATHWAY: 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.

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

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.

Attachment 4

STPEGS Emergency Action Level Deviation, Difference and Justification Matrix – revisions only

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

NEI 99-01 Rev. 6 Implementation

JUNE 2015

STPEGS EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

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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.

STP EAL DEVIATION/DIFFERENCE/JUSTIFICATION MATRIX

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**		
3. RCS Activity / Containment Radiation	A. Containment radiation monitor reading greater than (site-specific value). **OR**	Not Applicable	A 1.RCB Rad Monitor RT-8050 or RT-8051 greater than 2100 R/hr **OR** 2. HATCH MONITOR greater than 4200 mR/hr **OR**	Not Applicable	Difference	Loss A.- See Global Comment #9 Loss A.1 - Calc 15-RA-011 replaced STPNOC013-004 Rev. 2 due to consequential errors. Loss A.2 – Calc-03-ZE-003 revised to maintain consistency with Calc 15-RA-011
	B. (Site-specific indications that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131).		B. Sample analysis indicates that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131.		Difference	Loss B.- See Global Comment #9

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**		
3. RCS Activity / Containment Radiation	A. Containment radiation monitor reading greater than (site-specific value).	Not Applicable	A 1. RCB Rad Monitor RT-8050 or RT-8051 greater than 10 R/hr OR 2. HATCH MONITOR greater than 20 mR/hr	Not Applicable	Difference	Loss A.1 - Calc 15-RA-011 replaced STPNOC013-004 Rev. 2 due to consequential errors. Loss A.2 - New calculated value allows use of HATCH MONITOR as backup.

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**		
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 45,000 R/hr **OR** 2. HATCH MONITOR greater than 90,000 mR/hr	Difference	See Global Comment #9 Loss A.1 - Calc 15-RA-011 replaced STPNOC013-004 Rev. 2 due to consequential errors. Loss A.2 – Calc-03-ZE-003 revised to maintain consistency with Calc 15-RA-011.

Attachment 5

Support Documents

34125252

CALCULATION COVER SHEET

Calculation No.: 03-ZE-003 **Unit:** 9 **Bldg/Area/Sys:** MEAB & RCB / PAL / RA

Quality Class: Q **Priority Code:** 2

[ ] **Design Calculation** [X] **Engineering Calculation** **Cog. Org.:** **NFAD / Reactor Analysis**

Title: RT8050/RT8051 Contingency Conversion Constant for Post-Accident Failed Fuel Monitoring

Additional Review: **Dept:** N / A **Signature:** **Date:**

Additional Review: **Dept:** N / A **Signature:** **Date:**

RPE Certification Required: [ ] Yes [X] No

RPE Signature: **Date:** **Registration No.:**

RPE Seal:

☐ **This calculation revision contains a change in the methodology as described in UFSAR Section ________Rev____.**

CR actions tracking documents impacted by this revision to the calculation:

13-10171-71 **13-10171-72**

13-10171-73 **13-10171-74**

Approval Signature PRINT/SIGN		Date	Rev	Revision Description
Originator (ESP Cert 9569)	N. J. Hall / Nathaniel J Hall	6-9-15	1	New Reference Source from 15-RA-011 per CR action 13-10171-67.
Checker (ESP Cert 9569)	M. A. Whitley / Michael A Whitley	6-9-15		
SE	Duane Gore / Duane Gore	6/9/15		

STI: 34125252

INDEX TO CALCULATION REVISIONS

REV	CHANGE DOC. NO.	DESCRIPTION OF CHANGES	AFFECTED SHEETS	MODIFIED SHEETS
0	N/A	Initial Issue.	-	-
1	13-10171-67	Prepared for CR Action 13-10171-67 due to impact from 15-RA-011 superseding previous Reference source term. Table F and subsequent calculation steps are updated for a range of four source terms. Table E values for I-130 and Xe-137 are also added to support new source terms..		

List Of Effective Pages

Page No.	Latest Rev.	Page No.	Latest Rev.	Page No.	Latest Rev.	Page No.	Latest Rev.	Page No.	Latest Rev.	Page No.	Latest Rev.
1	1	25	1								
2	1	26	1								
3	1	27	1								
4	1	28	1								
5	1	1 CD	0								
6	1										
7	0										
8	0										
9	0										
10	0										
11	0										
12	0										
13	0										
14	0										
15	0										
16	0										
17	0										
18	0										
19	0										
20	1										
21	1										
22	1										
23	1										
24	1										

Total Number of Calculation Pages: 28

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1.0 Introduction

Implementation of TSC-288 requires that a contingency plan be in place for post-accident monitoring in the event of a failure of RT8050 or RT8051.

2.0 Purpose and Scope

The purpose of this evaluation is to determine the post-accident radiation detector response directly outside the Personnel Air Lock (PAL). This response will be used to calculate a conversion constant to calculate the RT8050/RT8051 response given a measured value outside the PAL. This process will be used in post-accident procedures if RT8050/RT8051 is unavailable.

3.0 Summary of Results

Conversion constants were calculated for 4 different source term assumptions which span the range of anticipated accidents. A single best estimate conversion constant was developed to cover the range of assumed source terms. The conversion constant for using the contingency portable monitor in lieu of RT8050/RT8051 is:

RT-8050/8051 response = **500 X** portable monitor

4.0 Assumptions

1. The release mix only contains those isotopes that will be airborne in containment (I, Kr, and Xe isotopes).

2. The PAL cyclinder and doors are composed of 100% iron (Fe).

3. The sources used in Cases 1-4 of Table F adequately bound both sprayed and no-spray conditions. See Figure 3.

4. The density of iron is 8.0 g/cm^3. The density of air is 0.00122 g/cm^3 (MicroShield default value).

5. This evaluation assumes that the outside containment measurement will be taken 9 feet outside the PAL at ~71' elevation. This should correspond closely to hanging a portable monitor from the hand-rail directly across from the PAL.

6. The likely source is bounded by those used to calculate the response of RT8050/RT8051 in Reference 7 and the maximum credible accident release from Reference 12.

5.0 References

1.Microshield v.5.

2.MCNP v. 4C.

3.MC-5281, "Free and Sprayed Volumes Inside Containment," Rev. 2.

4.Whitten, Kenneth W., Gailey, Kenneth D, and Davis, Raymond E., "General Chemistry", Third Edition, Saunders College Publishing, 1988.

5.Drawing 3M019C4047, "Concrete Mechanical and Electrical Aux. Bldg. Plan at El. 60'-0" Unit 1 and 2", Rev. 10.

6.Drawing 14926-0011(2)01352-FPD, "General Arrangement Personnel Air Lock".

7.Calculation 15-RA-011 Rev. 0, "Fission Product Barrier Failure for Emergency Action Level Thresholds", STI: 34117183.

8.5A369NQ1010, "Shielding Design Criteria", Rev. 5.

9.Drawing 14926-011(1)-01391-EPD, "Personnel Air Lock: Reactor Door Detail".

10. Drawing 14926-011(1)-01392-EPD, "Personnel Air Lock: Outer Door Detail".

11. MicroShield v. 9.07 (I-130 and Xe-137 only).

12. 0450-0100004WN, Radiation Analysis Manual, Rev. 5, May 1997, STI: 30521679.

13. Work Instruction PMI-IC-RA-8400A, "Four Channel Area Monitor" Rev. 8, Effective April 9, 2015; STI: 34104690.

14. Instrument Setpoint List database entries for RT-8050 and RT-8051

6.0 Method of Analysis

MicroShield (Reference 1) is used to calculate the gamma energy spectrum for each isotope in the release mix. The gamma energy spectrums are used in conjunction with the activity of each isotope to calculate a source spectrum-weighting matrix. MCNP (Reference 2) is used to calculate the responses per source particle for specific gamma energies at a dose point inside the containment building and a dose point at about 9 feet outside the PAL. The total response at each dose point is the spectrum-weighted summation of the MCNP results. The conversion constant is the ratio of the dose point inside containment to the dose point outside the PAL.

7.0 Calculations

7.1 MCNP Input

Input Deck

The MCNP input deck is shown below. The same deck is used for all runs with the only exception being the initial energy of the particle.

MCNP Input Deck

```
PAL Streaming Dose at Door 0.2 MeV
1     1 -0.00122 10 -11 -21 18
2     3 -2.18      10 -11 -21 17 -18
3     0            10 -11 -21 -17
4     2 -8.0       21 -22 10 -11
5     2 -8.0       9  -10 -22
6     2 -8.0       11 -12 -22
7     3 -2.18      2  8 16 -19 27 28 -24
8     3 -2.18 (1 -2 7 -14 15 -20):(5 -6 7 -14 15 -20)
            :(1 -6 7 -8  15 -20):(1 -6 13 -14 15 -20)
            :(1 -6 7 -14 15 -16):(1 -6 7 -14 19 -20)
9     3 -2.18  (3 -4 25 -13 16 -19):(3 -5 25 -26 16 -19)
10    3 -2.18  (2 8 17 -18 22 -23):(2 8 -9 17 -18 -22 -23)
11    0        (2 8 16 -17 22 -23):(2 8 -9 16 -17 -22)
12    1 -0.00122 (2 8 18 -19 22 -23):(2 8 18 -9 -22)
13    1 -0.00122 4 -5 26 -13 16 -19
14    1 -0.00122 (2 -3 8 -13 24 22 16 -19):(12 -22 -3 -13)
                :(3 8 24 -25 -5 16 -19)
15    3 -2.18    22 -27 23 -24 9
16    3 -2.18    2 8 16 -19 27 23 -28
17    0 -1:6:-7:14:-15:20

1     px 320.04
2     px 335.28
3     px 2529.84
4     px 2560.32
5     px 2987.04
6     px 3002.28
7     py 1219.2
8     py 1234.44
9     1 py 2273.31
10    1 py 2277.76
11    1 py 2816.79
12    1 py 2819.33
13    py 3108.96
14    py 3124.20
15    pz -15.24
16    pz 0
17    pz 228.6
18    pz 243.84
19    pz 762.0
20    pz 777.24
21    1 c/y 0 368.3 175.26
22    1 c/y 0 368.3 187.96
23    cz 2286.0
24    cz 2407.92
25    py 2316.48
26    py 2346.96
```

MCNP Input Deck

```
27   1 c/y 0 368.3 203.20
28   cz 2392.68

mode p
*tr1 0 0 0 30 120 90 60 30 90 90 90 0 1
imp:p 1 1 0 1 1 1 1 1 1 0 1 1 1 1 1 0
phys:p 1j 1 1
sdef par=2 x=d1 y=d2 z=d3 cel=12 erg=0.2 dir=d4 vec=0.5 0.866 0
si1  335.28 1924.05
sp1  0 1
si2  1234.44 2261.28
sp2  0 1
si3  243.84 762.0
sp3  0 1
si4 -1 0 1
sp4 0 1 1
sb4 0 0.1 0.9
fc5 MeV-barn per cc
f5:p 1546.83 2679.18 335.28 30
fm5 1 1 -5 -6
fc15 MeV-barn per cc
f15:p 1097.29 1900.54 411.48 30
fm15 1 1 -5 -6
m1 7000 -0.756 8000 -0.231 18000 -0.013 $ dry air
m2 26000 1
m3 1000 -0.0115 6000 -0.0023 8000 -1.1187 11000 -0.0364 12000 -0.0050
       13000 -0.0773 14000 -0.7683 19000 -0.0299 20000 -0.1000 26000
       -0.032
ctme 5
```

Model Layout

A graphical representation of the MCNP input deck is shown by Figure 1. The figures were created using the plotting feature in MCNP. The graphics when compared to the design drawings (Ref. 5 and Ref. 6) show the general layout of the model. The numbers on the graphics are surface labels from the input deck. The dimension of each surface is calculated in the table that follows.

Figure 1

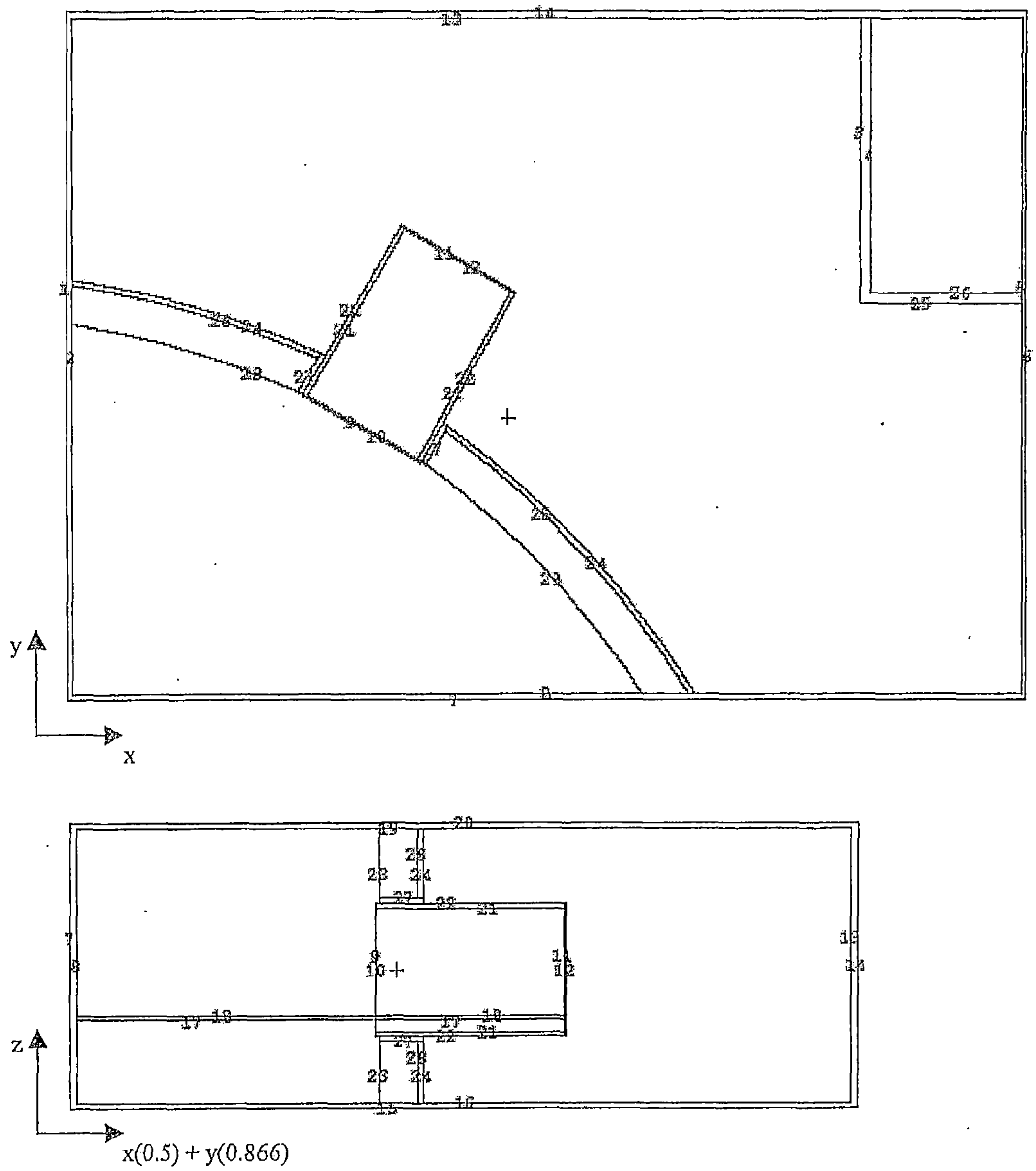


Table B contains the calculations for the MCNP surface dimension inputs. All dimensions are taken from References 5, 6, 9, and 10. The inner and outer radius of the containment building is 75' and 79' respectively. It should be noted that the dimensions are close approximations of design. Small discrepancies (± 3") on dimensions for walls will not have a significant impact on the results. The tolerance for the PAL doors is smaller because most of the exposure at the dose point is from radiation streaming through these doors.

Table B: MCNP Surface Dimensions

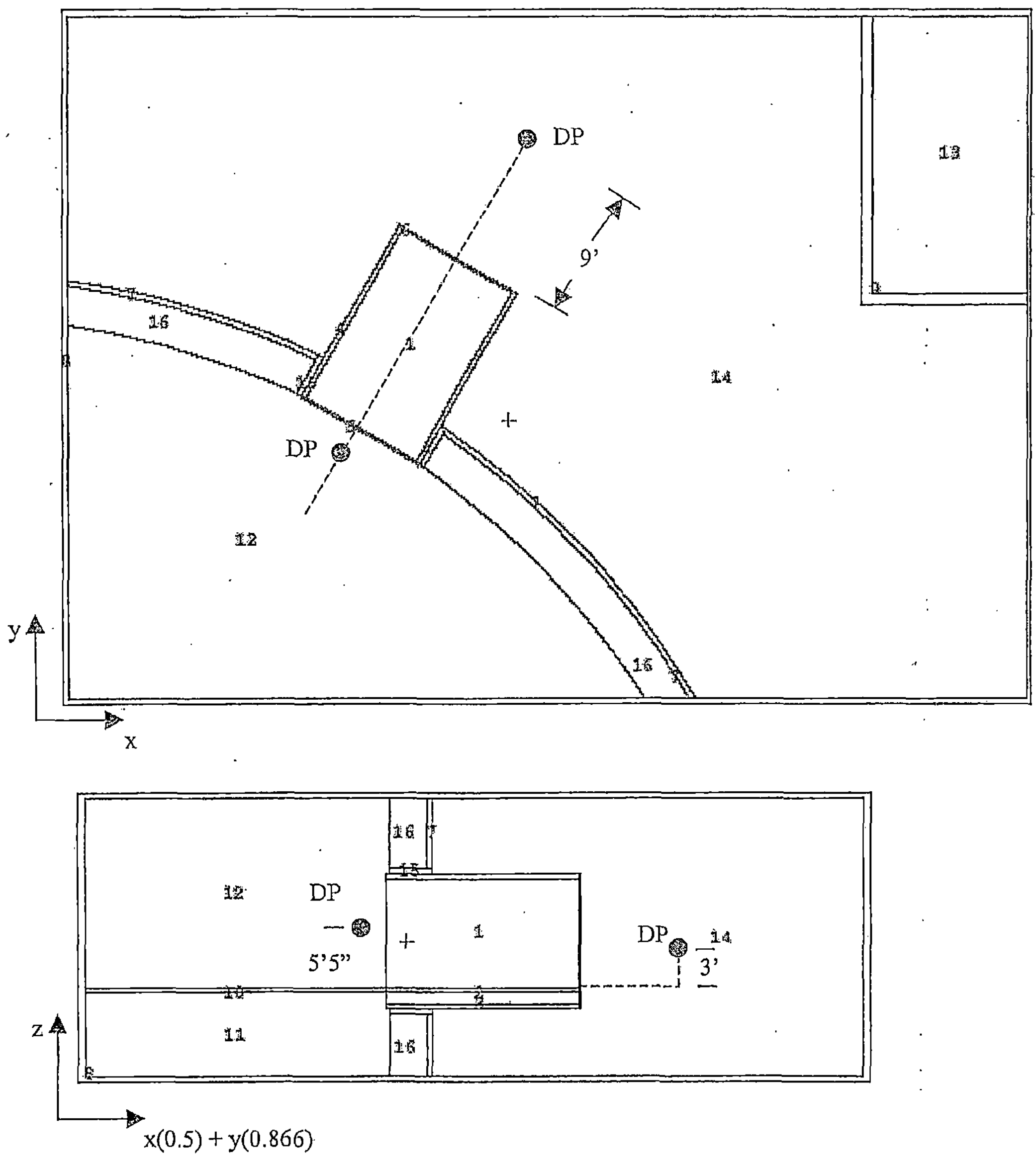
Surface	Type	Dimensions in Feet and Inches	Total in Centimeters
1	px	10' + 1' – 6"	320.04
2	px	10' + 1'	335.28
3	px	10' + 24' + 25' + 24'	2529.84
4	px	10' + 24' + 25' + 24' + 1'	2560.32
5	px	10' + 24' + 25' + 24' + 1' + 5.5" + 6'4.5" + 8'8" – 1'6"	2987.04
6	px	10' + 24' + 25' + 24' + 1' + 5.5" + 6'4.5" + 8'8" – 1'	3002.28
7	py	39' + 1'	1219.2
8	py	39' + 1'6"	1234.44
Note: Surfaces 9 through 12 are constructed using planes perpendicular to and cylinders parallel to the y-axis. These surfaces are translated 30 degrees to create the PAL. These dimensions are close approximations of the actual PAL.			
9	py	75' - ~5" This dimension is approximately 5 inches less than the inner radius of containment. This is necessary so that the edges of the door are not intersecting the containment wall in the model.	2273.31
10	py	75' - ~5" + 13/4"	2277.76
11	py	75' - ~5" + 13/4" + ~17'81/4"	2816.79
12	py	75' - ~5" + 2" + ~17'7" + 2"	2819.33
13	py	39' + 23' + 22' + 19' –1'	3108.96
14	py	39' + 23' + 22' + 19' –6"	3124.20
15	pz	Dimension is chosen as 6 inches below the 60' elevation for variance reduction purposes. The 60' elevation is the zero point on the z-axis.	-15.24
16	pz	This surface is the 60' foot elevation and is the zero point on the z-axis.	0
17	pz	8' – 6" This surface is 6 inches below the 68' elevation for variance reduction purposes.	228.6
18	pz	8'	243.84
19	pz	27' – 2' This surface is the ceiling of the model. It's 2 feet below the 87' elevation.	762.0
20	pz	27' – 1'6" This surface is 6 inches above the ceiling surface (19) for variance reduction purposes.	777.24
Surfaces 21 and 22 are cylinders parallel to the y-axis. The first two dimensions are the centerline offset from the x and z-axis. The third dimension is the radius of the cylinder.			

Table B: MCNP Surface Dimensions

21	c/y	x = 0 z = 12'1" (Reference 5) , radius = 5'9" (Reference 6)	0 368.3 175.26
22	c/y	x = 0 z = 12'1" (Reference 5) , radius = 6'2" (Reference 6)	0 368.3 187.96
23	cz	75', Containment Inner Radius	2286.0
24	cz	79', Containment Outer Radius	2407.92
25	py	39' + 23' + 22' – 8'	2316.48
26	py	39' + 23' + 22' – 8' + 1'	2346.96
27	c/y	x = 0 z = 12'1" (Reference 5), radius = 6'8" This surface is used to create a 6 inch slice of containment wall around the PAL for variance reduction.	0 368.3 203.20
28	cz	79' – 6" This surface is used to create a 6-inch slice of containment wall for variance reduction.	2392.68

Figure 2 shows the MCNP Input model with cell labels. The dose points (DP) from the model are added with dimensions.

Figure 2



* Outside cell is cell 17.

MCNP Card Descriptions

Photon Importance Cards (imp:p)

The photon importance of each cell is defined by the imp:p portion contained in each of the cell definition lines. The photon importance is set to zero in regions where there is a low probability for the photon to emerge from the region and contribute to the score. This increases the code efficiency and reduces the statistical variance.

Source Definition

The source is defined by the following line in the input deck.

```
sdef par=2 x=d1 y=d2 z=d3 cel=12 erg=0.2 dir=d4 vec=0.5 0.866 0
```

The source is a rectangular volume with x, y, and z ranges given by distribution d1, d2, and d3 respectively. The distributions are shown below.

```
si1  335.28 1924.05
sp1  0 1
si2  1234.44 2261.28
sp2  0 1
si3  243.84 762.0
sp3  0 1
```

The source definition above results in tracking only particles being born in cell 12. The remainder of the particles that are born in the rectangular volume are rejected.

The following lines are used to reduce the statistical variance by biasing the initial source particle direction. This biasing causes 90% of the sampled particles to be emitted with a positive cosine with respect to the reference vector. The reference vector, provided as vec = 0.5 0.866 0 in the sdef card, points in the direction parallel to the centerline of the PAL. This means that more particles are going in the direction of the dose point. This helps to reduce the statistical variance.

```
si4 -1 0 1
sp4 0 1 1
sb4 0 0.1 0.9
```

Point Detector

The dose point inside containment in the MCNP input is located 5'5" above the 68' elevation and 3' from the containment wall. The x-y coordinate of this point is assumed to be located on the line parallel to the PAL centerline. The coordinates for this point are calculated below.

$$R = 2286 - (3')(30.48) = 2194.56 \text{ cm}$$

The point is determined by solving for the intersection of the equation for the circle that goes through 2194.56 and the PAL centerline. The two equation are below.

$$y = x / \tan(\theta) \text{ or } x = y\tan(\theta), \quad \theta = 30° \text{ (Ref. 5)} \qquad x^2 + y^2 = 2194.56^2$$

Therefore, substituting into the circle equation yields the following expression.

$$y = \sqrt{\frac{2194.56^2}{\tan^2(\theta)+1}} = 1900.54 \text{ cm} \qquad \text{and } x = 1900.54\tan(30) = 1097.29 \text{ cm}$$

$$z = 8' + 5'5" = 13.5'(30.48) = 411.48 \text{ cm}$$

The dose point outside containment in the MCNP input is located 9 feet from the outside of the PAL door on the MEAB side. The coordinates for the dose point are calculated by using surface 12. Surface 12 is a plane parallel to the y-axis that has been rotated 30 degrees. The x and y dimension on the centerline on the outside of the PAL door calculated below.

$$x = 2819.33 \sin(30) = 1409.67 \text{ cm}$$
$$y = 2819.33 \cos(30) = 2441.61 \text{ cm}$$

The x and y coordinate of the dose point is determined by calculating the x and y coordinates of a point 9 feet away from the coordinates calculated above.

$$x = 1409.67 + (9')(30.48) \sin(30) = 1546.83 \text{ cm}$$
$$y = 2441.61 + (9')(30.48) \cos(30) = 2679.18 \text{ cm}$$

The detector is located 3 feet above the 68' elevation.

$$z = 243.84 + (3')(30.48) = 335.28 \text{ cm}$$

The detector is given a 30 cm sphere of exclusion.

Material Definitions

The MCNP model uses three materials compositions. Concrete for the walls, iron for the PAL cylinder and doors, and air in the rooms.

The constituent densities of concrete are listed in Table C

Table C: Concrete Data, $\rho = 2.18 \frac{gm}{cm^3}$ (Ref. 8, Pg. 10)					
Constituent	Density (g/cm^3)	Constituent	Density (g/cm^3)	Constituent	Density (g/cm^3)
H	0.0115	Mg	0.005	Ca	0.1
C	0.0023	Al	0.0773	Fe	0.0317
O	1.1187	Si	0.7683		
Na	0.0364	K	0.0299		

The PAL cylinders and doors are composed of 100% iron. (Assumption 2)

The constituent densities of air are calculated. The air is assumed to consist only of Nitrogen, Oxygen, and Argon. The molecular weight and relative volume composition of each constituent is listed in Table D.

Table D: Air Data (Ref. 4)		
Constituent	Molecular Weight (g/mol)	Relative Volume Composition
N_2	28.01348	78.09
O_2	31.9988	20.94
Ar	39.948	0.93
	Ref. 4	Ref. 4, Pg. 261

The following equations are used to determine the constituent densities of air.

$$\frac{n_{N_2} MW_{N_2}}{V} + \frac{n_{O_2} MW_{O_2}}{V} + \frac{n_{Ar} MW_{Ar}}{V} = \rho_{Air}$$ Equation 1

$$n_{N_2} = \frac{RVC_{N_2}}{RVC_{O_2}} n_{O_2}$$ Equation 2

$$n_{Ar} = \frac{RVC_{Ar}}{RVC_{O_2}} n_{O_2}$$ Equation 3

n_i = number of moles of molecule i
MW_i = molecular weight of molecule i
RVC_i = relative volume composition of molecule i
V = volume (1 cm^3)
ρ_{Air} = density of air (0.00122 g/cm^3) (Assumption 2)

The three equations above are used to get the expressions for the relative density of each constituent.

$$\rho_{N_2} = \left[\frac{\rho_{Air}}{MW_{N_2} + \frac{RVC_{O_2}}{RVC_{N_2}} MW_{O_2} + \frac{RVC_{Ar}}{RVC_{N_2}} MW_{Ar}} \right] MW_{N_2} \qquad \text{Equation 4}$$

$$\rho_{O_2} = \left[\frac{\rho_{Air}}{\frac{RVC_{N_2}}{RVC_{O_2}} MW_{N_2} + MW_{O_2} + \frac{RVC_{Ar}}{RVC_{O_2}} MW_{Ar}} \right] MW_{O_2} \qquad \text{Equation 5}$$

$$\rho_{AR} = \left[\frac{\rho_{Air}}{\frac{RVC_{N_2}}{RVC_{Ar}} MW_{N_2} + \frac{RVC_{O_2}}{RVC_{Ar}} MW_{O_2} + MW_{Ar}} \right] MW_{Ar} \qquad \text{Equation 6}$$

Applying the data from Table D to Equations 4, 5, and 6 yields the following constituent densities.

$$\rho_{N_2} = 0.000922 \ \frac{\text{gm}}{\text{cm}^3} \qquad \rho_{O_2} = 0.000282 \ \frac{\text{gm}}{\text{cm}^3} \qquad \rho_{Ar} = 0.000016 \ \frac{\text{gm}}{\text{cm}^3}$$

The weight percent composition of each constituent in dry air is as follows.

Nitrogen = 0.000922 / 0.00122 × 100% = 75.6%
Oxygen = 0.000282 / 0.00122 × 100% = 23.1%
Argon = 0.000016 / 0.00122 × 100% = 1.3%

7.2 Gamma Emission Rate

Table E contains the gamma emission rate per Curie (Ci) of each isotope in the release mix. The gamma emission rate per Curie is taken from MicroShield (Reference 1). The emission rate is determined by creating a source of 1 Ci for each isotope and displaying the gamma energy spectrum.

Table E: Gamma Emission Rates
(Gamma/sec-Ci)

Energy (MeV)	I-131	I-132	I-133	I-134	I-135	Kr-83m
0.2	9.7991E+07	5.1127E+07	0.0000E+00	1.0838E+09	7.3147E+08	0.0000E+00
0.3	2.4173E+09	9.2027E+08	2.1335E+08	2.3899E+08	1.3235E+09	0.0000E+00
0.4	3.0031E+10	8.6915E+08	2.1239E+08	4.6775E+09	1.8842E+09	0.0000E+00
0.5	1.3339E+08	8.7864E+09	3.2834E+10	4.8963E+09	2.7523E+09	0.0000E+00
0.6	2.7676E+09	4.7270E+10	4.3819E+08	1.2539E+10	2.6359E+08	0.0000E+00
0.8	6.6692E+08	3.5080E+10	2.8895E+09	6.5103E+10	2.8295E+09	0.0000E+00
1	0.0000E+00	1.0841E+10	8.8596E+08	1.4273E+10	1.5321E+10	0.0000E+00
1.5	0.0000E+00	5.7069E+09	9.1726E+08	4.9669E+09	2.0036E+10	0.0000E+00
2	0.0000E+00	1.2161E+09	0.0000E+00	2.9650E+09	4.0713E+09	0.0000E+00
3	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
4	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00

Table E: Gamma Emission Rates (continued)
(Gamma/sec-Ci)

Energy (MeV)	Kr-85m	Kr-85	Kr-87	Kr-88	Kr-89	Xe-131m
0.2	0.0000E+00	0.0000E+00	0.0000E+00	9.7079E+09	8.2007E+09	0.0000E+00
0.3	5.1759E+09	0.0000E+00	0.0000E+00	9.3455E+07	8.0734E+08	0.0000E+00
0.4	0.0000E+00	0.0000E+00	1.8315E+10	1.1176E+09	3.7955E+09	0.0000E+00
0.5	0.0000E+00	1.6058E+08	0.0000E+00	2.6884E+08	3.2937E+09	0.0000E+00
0.6	7.7989E+06	0.0000E+00	7.0515E+08	8.7054E+07	9.4624E+09	0.0000E+00
0.8	0.0000E+00	0.0000E+00	3.0330E+09	5.3564E+09	6.1250E+09	0.0000E+00
1	0.0000E+00	0.0000E+00	4.6705E+08	2.9150E+09	7.0537E+09	0.0000E+00
1.5	0.0000E+00	0.0000E+00	1.6086E+09	6.4587E+09	1.0956E+10	0.0000E+00
2	0.0000E+00	0.0000E+00	1.2014E+09	2.2496E+10	6.0169E+09	0.0000E+00
3	0.0000E+00	0.0000E+00	5.1740E+09	2.8548E+08	3.1857E+09	0.0000E+00
4	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	1.6569E+09	0.0000E+00

Table E: Gamma Emission Rates (continued) (Gamma/sec-Ci)					
Energy (MeV)	Xe-133m	Xe-133	Xe-135m	Xe-135	Xe-138
0.2	3.8110E+09	2.6337E+07	0.0000E+00	3.3263E+10	1.2937E+09
0.3	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	1.1853E+10
0.4	0.0000E+00	0.0000E+00	0.0000E+00	2.1388E+08	1.0838E+10
0.5	0.0000E+00	0.0000E+00	2.9969E+10	0.0000E+00	2.7040E+08
0.6	0.0000E+00	0.0000E+00	0.0000E+00	1.1485E+09	2.5524E+08
0.8	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	3.8811E+08
1	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	2.6352E+09
1.5	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	1.8531E+08
2	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	1.5295E+10
3	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
4	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00

7.3 Source Spectrum Weighting

The source spectrum-weighting factors are calculated using the relative abundance of each energy to the total energy spectrum. The weighting factor is the summation of the product of the activity of each isotope and the gamma emission rate per Ci at that energy.

The energy dependent weighting factors (W(E))are calculated using the Equation 7 below.

$$W(E) = \sum_{i=1}^{\#isotopes} Src_i \times \text{Table } E_i \qquad \text{Equation 7}$$

where, Src_i = activity of isoptope i from Table F

Table E_i = γ/sec-Ci for isotope i from Table E

The post-accident exposure to RT8050/RT8051 for 3 separate cases is calculated in Reference 7. A separate case, the maximum credible accident release, is taken directly from Reference 12, Table 5-9. The source term used in that evaluation is used here to generate the weighting factors. The relative abundance of each isotope changes depending on the assumed source term. This results in a different conversion factor for each source term. In order to accommodate the new source terms, two isotopes (I-130 and Xe-137) which were previously excluded from Table E had to be added. Data for these two isotopes was calculated with Microshield 9.07 (Reference 11), and is included in Table E1 below. Table F provides the reference sources.

Table E1: Gamma Emission Rates for I-130 and Xe-137 (Gamma / sec - Ci)

Energy (MeV)	I-130	Xe137
0.2	0.0000E+00	0.0000E+00
0.3	0.0000E+00	4.3164E+07
0.4	1.2637E+10	5.1115E+07
0.5	3.7549E+10	1.1359E+10
0.6	3.7062E+10	0.0000E+00
0.8	3.0951E+10	2.2718E+08
1	4.8722E+09	1.1507E+08
1.5	4.0440E+08	6.5239E+08
2	0.0000E+00	1.5107E+08
3	0.0000E+00	6.7018E+07
4	0.0000E+00	0.0000E+00

Table F: Reference Sources

Isotope	Case 1 (Ref. 7)	Case 2 (Ref. 7)	Case 3 (Ref. 7)	Case 4 (Ref. 12)
I-130		8.60E+03	8.60E+04	2.10E+06
I-131	1.79E+02	5.49E+04	2.20E+06	5.30E+07
I-132	2.53E+02	7.49E+04	3.00E+06	7.60E+07
I-133	2.96E+02	1.10E+05	4.40E+06	1.10E+08
I-134	5.49E+01	1.20E+05	4.80E+06	1.20E+08
I-135	8.02E+02	9.98E+04	4.00E+06	1.00E+08
Kr83m	9.81E+01	2.60E+04	2.60E+05	1.40E+07
Kr85	2.01E+03	7.40E+03	7.40E+04	1.20E+06
Kr85m	3.98E+02	5.80E+04	5.80E+05	2.90E+07
Kr87	2.60E+02	1.10E+05	1.10E+06	5.50E+07
Kr88	7.42E+02	1.56E+05	1.56E+06	7.80E+07
Kr89	2.23E+01	1.90E+05	1.90E+06	9.50E+07
Xe131m	7.42E+02	2.20E+03	2.20E+04	1.10E+06
Xe133	6.36E+04	4.40E+05	4.40E+06	2.20E+08
Xe133m	1.11E+03	1.36E+04	1.36E+05	6.80E+06
Xe135	2.01E+03	1.10E+05	1.10E+06	5.50E+07
Xe135m	1.06E+02	8.40E+04	8.40E+05	4.20E+07
Xe137	4.24E+01	3.80E+05	3.80E+06	1.90E+08
Xe138	1.54E+02	3.60E+05	3.60E+06	1.80E+08

The weighting factors calculated by Equation 7 are presented in Table G.

Table G: Energy Specific Weighting Factors

Energy (MeV)	Case 1 Weight Factor	Case 2 Weight Factor	Case 3 Weight Factor	Case 4 Weight Factor
0.2	8.100E+13	7.473E+15	8.110E+16	3.843E+18
0.3	5.778E+12	5.138E+15	6.298E+16	2.758E+18
0.4	1.520E+13	9.450E+15	1.692E+17	5.885E+18
0.5	1.874E+13	1.306E+16	2.849E+17	9.028E+18
0.6	1.629E+13	7.698E+15	2.353E+17	6.450E+18
0.8	2.066E+13	1.390E+16	4.715E+17	1.246E+19
1	1.893E+13	7.032E+15	1.950E+17	5.596E+18
1.5	2.357E+13	6.708E+15	1.610E+17	4.925E+18
2	2.324E+13	1.120E+16	1.376E+17	6.030E+18
3	1.629E+12	1.244E+15	1.244E+16	6.222E+17
4	3.695E+10	3.148E+14	3.148E+15	1.574E+17

7.4 MCNP Results / Conversion Calculations

The MCNP runs (input/output) are contained in the CD attached to this document for Revision 0. The following files remain applicable for Revision 1.

Input	Output
palstream_wfloor_0.2mev_16to1	palstream_wfloor_0.2mev_16to1.o
palstream_wfloor_0.3mev_16to1	palstream_wfloor_0.3mev_16to1.o
palstream_wfloor_0.4mev_16to1	palstream_wfloor_0.4mev_16to1.o
palstream_wfloor_0.5mev_16to1	palstream_wfloor_0.5mev_16to1.o
palstream_wfloor_0.6mev_16to1	palstream_wfloor_0.6mev_16to1.o
palstream_wfloor_0.8mev_16to1	palstream_wfloor_0.8mev_16to1.o
palstream_wfloor_1mev_16to1	palstream_wfloor_1mev_16to1.o
palstream_wfloor_1.5mev_16to1	palstream_wfloor_1.5mev_16to1.o
palstream_wfloor_2mev_16to1	palstream_wfloor_2mev_16to1.o
palstream_wfloor_3mev_16to1	palstream_wfloor_3mev_16to1.o
palstream_wfloor_4mev_16to1	palstream_wfloor_4mev_16to1.o

Tables H1 – H4 (on the following pages) contain the results of the 11 MCNP runs, and show the calculation of the total weighted response calculation for each dose point for each case.

Table H1: Total Detector Response - Case 1

A	B	C	D	E	F
		Inside Cont. DP		Outside Cont. DP	
Energy (MeV)	Weighting Factor	MCNP Result	Weighted Response	MCNP Result	Weighted Response
0.2	8.100E+13	1.0676E-07	8.650E+06	2.9048E-12	2.353E+02
0.3	5.778E+12	1.6756E-07	9.682E+05	2.7440E-11	1.585E+02
0.4	1.520E+13	2.2342E-07	3.396E+06	8.1880E-11	1.245E+03
0.5	1.874E+13	2.7512E-07	5.160E+06	1.6311E-10	3.060E+03
0.6	1.629E+13	3.2329E-07	5.266E+06	2.7033E-10	4.404E+03
0.8	2.066E+13	4.1164E-07	8.504E+06	5.5488E-10	1.146E+04
1	1.893E+13	4.9031E-07	9.282E+06	9.0160E-10	1.707E+04
1.5	2.357E+13	6.5599E-07	1.546E+07	2.0002E-09	4.714E+04
2	2.324E+13	7.9747E-07	1.853E+07	3.0696E-09	7.134E+04
3	1.629E+12	1.0403E-06	1.695E+06	5.1829E-09	8.443E+03
4	3.695E+10	1.2575E-06	4.646E+04	6.8990E-09	2.549E+02
		Total	7.700E+07	Total	1.648E+05
			= B x C		= B x E

Case 1 Inside / Outside Ratio = **467.2**

Table H2: Total Detector Response - Case 2

A	B	C	D	E	F
		Inside Cont. DP		Outside Cont. DP	
Energy (MeV)	Weighting Factor	MCNP Result	Weighted Response	MCNP Result	Weighted Response
0.2	7.473E+15	1.0676E-07	7.978E+08	2.9048E-12	2.171E+04
0.3	5.138E+15	1.6756E-07	8.609E+08	2.7440E-11	1.410E+05
0.4	9.450E+15	2.2342E-07	2.111E+09	8.1880E-11	7.738E+05
0.5	1.306E+16	2.7512E-07	3.593E+09	1.6311E-10	2.130E+06
0.6	7.698E+15	3.2329E-07	2.489E+09	2.7033E-10	2.081E+06
0.8	1.390E+16	4.1164E-07	5.722E+09	5.5488E-10	7.713E+06
1	7.032E+15	4.9031E-07	3.448E+09	9.0160E-10	6.340E+06
1.5	6.708E+15	6.5599E-07	4.400E+09	2.0002E-09	1.342E+07
2	1.120E+16	7.9747E-07	8.932E+09	3.0696E-09	3.438E+07
3	1.244E+15	1.0403E-06	1.294E+09	5.1829E-09	6.448E+06
4	3.148E+14	1.2575E-06	3.959E+08	6.8990E-09	2.172E+06
		Total	3.404E+10	Total	7.562E+07
			= B x C		= B x E

Case 2 Inside / Outside Ratio = **450.1**

Table H3: Total Detector Response - Case 3

A	B	C	D	E	F
		Inside Cont. DP		Outside Cont. DP	
Energy (MeV)	Weighting Factor	MCNP Result	Weighted Response	MCNP Result	Weighted Response
0.2	8.110E+16	1.0676E-07	8.658E+09	2.9048E-12	2.356E+05
0.3	6.298E+16	1.6756E-07	1.055E+10	2.7440E-11	1.728E+06
0.4	1.692E+17	2.2342E-07	3.780E+10	8.1880E-11	1.385E+07
0.5	2.849E+17	2.7512E-07	7.838E+10	1.6311E-10	4.647E+07
0.6	2.353E+17	3.2329E-07	7.607E+10	2.7033E-10	6.361E+07
0.8	4.715E+17	4.1164E-07	1.941E+11	5.5488E-10	2.616E+08
1	1.950E+17	4.9031E-07	9.561E+10	9.0160E-10	1.758E+08
1.5	1.610E+17	6.5599E-07	1.056E+11	2.0002E-09	3.220E+08
2	1.376E+17	7.9747E-07	1.097E+11	3.0696E-09	4.224E+08
3	1.244E+16	1.0403E-06	1.295E+10	5.1829E-09	6.450E+07
4	3.148E+15	1.2575E-06	3.959E+09	6.8990E-09	2.172E+07
		Total	7.334E+11	Total	1.394E+09
			= B x C		= B x E

Case 3 Inside / Outside Ratio = **526.1**

Table H4: Total Detector Response - Case 4

A	B	C	D	E	F
		Inside Cont. DP		Outside Cont. DP	
Energy (MeV)	Weighting Factor	MCNP Result	Weighted Response	MCNP Result	Weighted Response
0.2	3.843E+18	1.0676E-07	4.103E+11	2.9048E-12	1.116E+07
0.3	2.758E+18	1.6756E-07	4.621E+11	2.7440E-11	7.568E+07
0.4	5.885E+18	2.2342E-07	1.315E+12	8.1880E-11	4.819E+08
0.5	9.028E+18	2.7512E-07	2.484E+12	1.6311E-10	1.473E+09
0.6	6.450E+18	3.2329E-07	2.085E+12	2.7033E-10	1.744E+09
0.8	1.246E+19	4.1164E-07	5.129E+12	5.5488E-10	6.914E+09
1	5.596E+18	4.9031E-07	2.744E+12	9.0160E-10	5.045E+09
1.5	4.925E+18	6.5599E-07	3.231E+12	2.0002E-09	9.851E+09
2	6.030E+18	7.9747E-07	4.809E+12	3.0696E-09	1.851E+10
3	6.222E+17	1.0403E-06	6.473E+11	5.1829E-09	3.225E+09
4	1.574E+17	1.2575E-06	1.979E+11	6.8990E-09	1.086E+09
		Total	2.351E+13	Total	4.842E+10
			= B x C		= B x E

Case 4 Inside / Outside Ratio = **485.5**

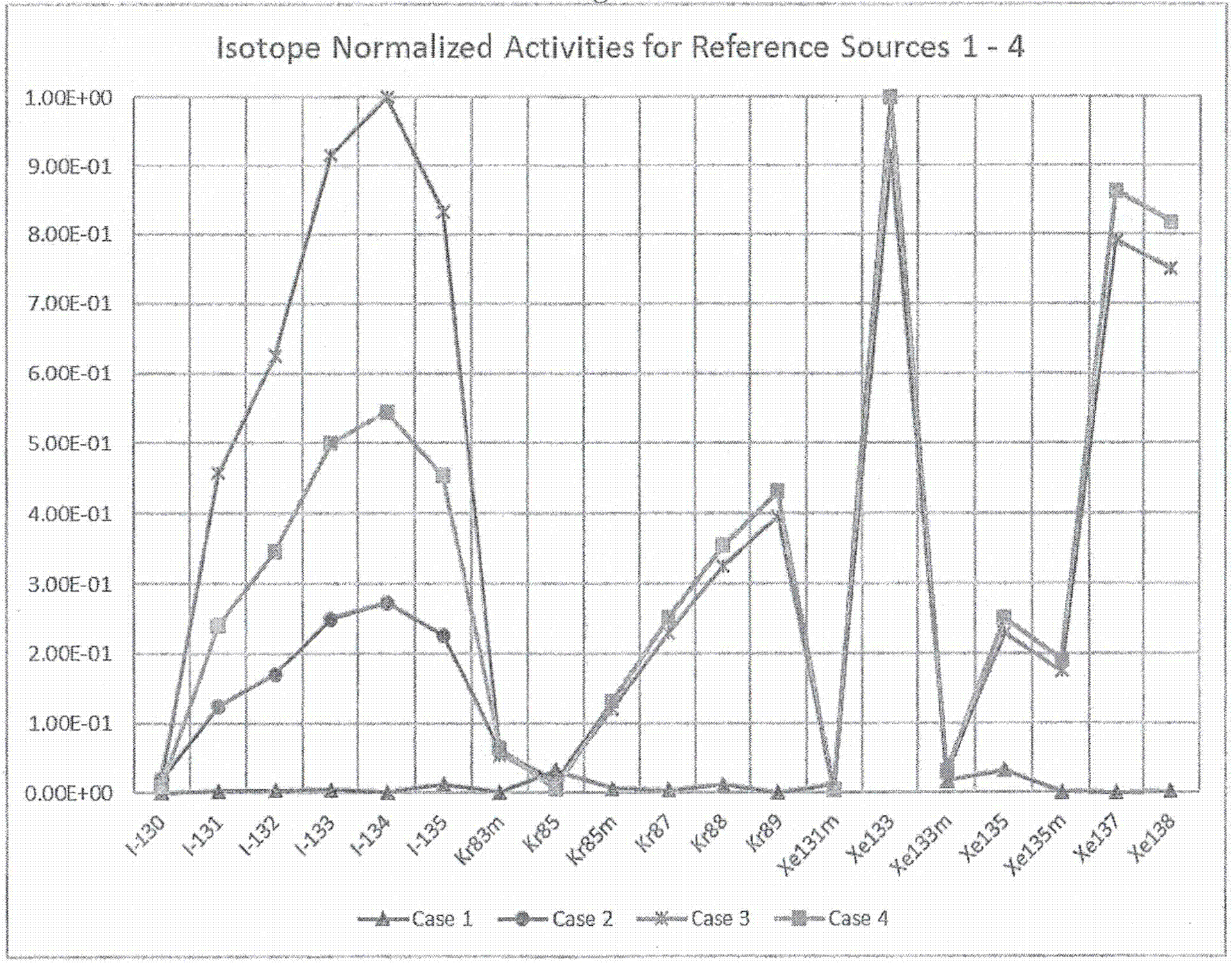
Although, no containment spray is considered in the four source terms considered, they represent a wide range of accident scenarios with a wide range of normalized activities for the various isotopes. There is an especially wide variance with the Iodine isotopes. To calculate the normalized activities for each source, each isotope's activity is divided by the maximum activity of that source. Table I shows the normalized activities for each reference source (Cases 1 – 4).

Table I: Reference Source Isotope Relative Activities

Isotope	Case 1	Case 2	Case 3	Case 4
I-130	0.00E+00	1.95E-02	1.79E-02	9.55E-03
I-131	2.81E-03	1.25E-01	4.58E-01	2.41E-01
I-132	3.98E-03	1.70E-01	6.25E-01	3.45E-01
I-133	4.65E-03	2.50E-01	9.17E-01	5.00E-01
I-134	8.63E-04	2.73E-01	1.00E+00	5.45E-01
I-135	1.26E-02	2.27E-01	8.33E-01	4.55E-01
Kr83m	1.54E-03	5.91E-02	5.42E-02	6.36E-02
Kr85	3.16E-02	1.68E-02	1.54E-02	5.45E-03
Kr85m	6.26E-03	1.32E-01	1.21E-01	1.32E-01
Kr87	4.09E-03	2.50E-01	2.29E-01	2.50E-01
Kr88	1.17E-02	3.55E-01	3.25E-01	3.55E-01
Kr89	3.51E-04	4.32E-01	3.96E-01	4.32E-01
Xe131m	1.17E-02	5.00E-03	4.58E-03	5.00E-03
Xe133	1.00E+00	1.00E+00	9.17E-01	1.00E+00
Xe133m	1.75E-02	3.09E-02	2.83E-02	3.09E-02
Xe135	3.16E-02	2.50E-01	2.29E-01	2.50E-01
Xe135m	1.67E-03	1.91E-01	1.75E-01	1.91E-01
Xe137	6.67E-04	8.64E-01	7.92E-01	8.64E-01
Xe138	2.42E-03	8.18E-01	7.50E-01	8.18E-01

Figure 3 shows the information from Table I in graphical form.

Figure 3



Note that Case 1 has almost no Iodines relative to the Xe-133 inventory. This is due to the release nature of a typical small defect fuel failure that may occur during normal operations. Case 1 assumes a failure of the RCS boundary with no accident related fuel failures. Cases 2 – 4 involve accident scenarios with accident related cladding failures which cause a rapid release of the nuclides in the fuel rod gap. This increases the Iodine and short-lived noble gas normalized activities. Figure 3 shows that a wide range of normalized inventories are represented by Cases 1 – 4. Therefore the results for the conversion constants can be reasonably expected to be bounding for anticipated scenarios. However, since the release characteristics will not be known during an accident, the results for Cases 1 – 4 should be used to develop one conversion constant useful for a range of release types.

Since it is anticipated that these results will be used in developing EAL thresholds and procedures for a portable monitor outside the PAL door, it is advantageous to have one conversion constant. It is not beneficial to significantly over-estimate or under-estimate the dose rate in containment. Therefore a best estimate value is desirable.

The conversion constant of 500 is selected to minimize under-estimation of the containment dose rate for Cases 3 while maintaining the relative error of all cases within the +/- 15 % tolerance limit for RT-8050 and RT-8051 (Reference 14). Table J below documents the relative errors for each case based on the single constant.

Table J: Single Conversion Constant for PAL Door to In-Containment Exposure Rates

A	B	C	D
Source Term	Conversion Factor	Single Factor	% Error
Case 1	467.2	500	-6.6%
Case 2	450.1		-10.0%
Case 3	526.1		5.2%
Case 4	485.5		-2.9%
	See Tables H1 – H4	Best Estimate	1 – (B ÷ C)

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Form 3	Calculation Impact Assessment	Page 1 of 1

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Inter-discipline Coordination Required? ☐ YES ☒ NO

		Initial/Date			Initial/Date
☐	Risk and Reliability Analysis		☐	Thermal Hydraulics (Ch 15 Safety Analysis, HELBA)	
☐	Reactor Engineering (including Radiological Impact)		☐	Civil Engineering	
☐	Mechanical Engineering		☐	Electrical Engineering	
☐	Equipment Qualification		☐	I&C Engineering	
			☐	Other:	

List sections reviewed of applicable documents/applicable attributes of the calculation/CRs generated:
This calculation was prepared in support of a Emergency Plan revision, so impacts are anticipated. The entire Emergency Plan is being updated to the guidance in Reference 1 (NEI 99-01). See CR 13-10171.

TECHNICAL SPECIFICATIONS SECTIONS REVIEWED/IMPACT:
None

UFSAR SECTIONS REVIEWED/IMPACT:
Sections 11.5 and 12.3 / None

SER SECTIONS REVIEWED/IMPACT:
None

DESIGN BASIS DOCUMENTS REVIEWED/IMPACT:
None

PROCEDURES REVIEWED/IMPACT:
0ERP01-ZV-IN01 / EAL-5, and various Initiating Conditions; 0PEP02-ZG-0007 / Form 1 Step 2.1.2.2 requires updated; 0PRP11-ZR-0006 / Step 4.2 requires update; 0POP04-RA-0001 / Addendum 27 requires update

OTHER CALCULATIONS REVIEWED/IMPACT:
12-RA-003 / None

OTHER DOCUMENTS REVIEWED/IMPACT:
Emergency Plan impacts are anticipated and will be implemented by action 13-10171-66.

certrec.com FILES SEARCHED AND SEARCH PARAMETERS USED:
UFSAR 11.5 and 12.3:"8050", "portable", "handheld", "hatch", "airlock", "PAL", "containment"

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Preparation of Calculations		
Form 4	Checkers Comments	Page 1 of 1

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The Following attributes are required by this procedure

Yes	N/A	
☒		The calculation has been assigned a Priority Code (3.1.2)
☒		The calculation has been assigned a Quality Class (3.1.3)
☒		The calculation contains the pages identified in (3.1.4)
☒		The calculation contains an Objective and Scope (3.1.5.1)
☒		The calculation contains a Summary of Results (3.1.5.2)
☒		The calculation describes the Method of Analysis (3.1.5.3)
☒		The calculation correctly identifies inputs (3.1.5.4)
☐	☒	A CR action has been issued to track inputs that require confirmation (3.1.5.4)
☒		The calculation identifies the references (3.1.5.5)
☒		The calculation clearly identifies assumptions (3.1.5.6)
☐	☒	A CR action has been issued to track assumptions that require confirmation (3.1.5.6)
☒		The calculation adequately presents the results (3.1.5.7)
☐	☒	Computer analysis meets the requirements of 3.1.5.8
☒		The calculation contains an impact review (3.1.5.9)
☒		The Originator and Checker are qualified to Engineering Support Certification 9569, "Design and Engineering Calculations" as shown in the Qual King database.

Checker's calculation summary

The assumptions, inputs, methods, calculations and conclusions were reviewed. The calculation presents an accurate best estimate conversion factor applicable to a wide range of postulated source terms.

CALCULATION COVER SHEET

Calculation No.: 15-RA-011 **Unit:** 9 **Bldg/Area/Sys:** RCB / 68' / RA

Quality Class: Q **Priority Code:** 2

[ ] **Design Calculation** [X] **Engineering Calculation** **Cog. Org.:** **NFAD / Reactor Analysis**

Title: Fission Product Barrier Failure for Emergency Action Level Thresholds

Additional Review: **Dept:** N / A **Signature:** **Date:**

Additional Review: **Dept:** N / A **Signature:** **Date:**

RPE Certification Required: [ ] **Yes** [X] **No**

RPE Signature: **Date:** **Registration No.:**

RPE Seal:

☐ **This calculation revision contains a change in the methodology as described in UFSAR Section ________ Rev____.**

CR actions tracking documents impacted by this revision to the calculation:

13-10171-67

Approval Signature PRINT/SIGN		Date	Rev	Revision Description
Originator (ESP Cert 9569)	N. J. Hall / Nathaniel J. Hall	5-26-15	0	Initial Issue, See CR Action 15-10595-1. Supersedes STPNOC013-CALC-004 (STI: 34061762). Supersedes results for RT-8050/8051 from 91-RA-001 (STI: 31338218)
Checker (ESP Cert 9569)	M. A. Whitley / [signature]	5-26-15		
SE	Duane Gore / [signature]	5/27/15		

STI: 34117183

INDEX TO CALCULATION REVISIONS

REV	CHANGE DOC. NO.	DESCRIPTION OF CHANGES	AFFECTED SHEETS	MODIFIED SHEETS
0	N/A	Initial Issue.. Prepared for CR Action 15-10595-1.	-	-

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List Of Effective Pages

Page No.	Latest Rev.	Page No.	Latest Rev.	Page No.	Latest Rev.	Page No.	Latest Rev.	Page No.	Latest Rev.	Page No.	Latest Rev.
1	0	B9	0								
2	0	B10	0								
3	0	1 CD	0								
4	0										
5	0										
6	0										
7	0										
8	0										
9	0										
10	0										
11	0										
12	0										
13	0										
14	0										
15	0										
A1	0										
B1	0										
B2	0										
B3	0										
B4	0										
B5	0										
B6	0										
B7	0										
B8	0										

Total Number of Calculation Pages: 26

TABLE OF CONTENTS

List of Tables

Objective and Scope

This calculation documents the predicted containment radiation monitor response for 3 postulated release scenarios patterned after the guidance in NEI-99-01 Rev. 6 (Reference 1). This calculation applies to both units. The subject detectors are RE-8050 and RE-8051.

Results for the following Emergency Action Level Thresholds are developed in this calculation:

Case 1: Loss of RCS Barrier Only – PWR RCS Barrier Threshold Loss 3.A
Case 2: Loss of RCS and Fuel Cladding Barrier – PWR Fuel Clad Barrier Threshold Loss 3.A
Case 3: Challenged Containment Barrier – PWR Containment Barrier Threshold Potential Loss 3.A

Summary of Results

See Appendix B for the detailed Microshield output. The results with buildup are reported below. They are rounded to the nearest R/hr.

Case 1 Exposure Rate:	**13 R/hr**
Case 2 Exposure Rate:	**2144 R/hr**
Case 3 Exposure Rate:	**45,040 R/hr**

Inputs

1. Containment Height = 183.9 ft (5605.27 cm) — Reference 5, Table 7.2.1 (See Assumption 2)
2. Containment Radius = 75 ft (2286.0 cm) — Reference 6
3. Containment Free Volume = 3.30 X 10^6 ft^3 — Reference 5, Table 7.2.1-1
4. Reactor Coolant Mass = 2.65 X 10^8 grams — Reference 3, Table 5-15
5. RCS 1% Failure Specific Activity $\bar{E}$ = .36 MeV/Dis — Reference 3, page 5-26
6. Density of Air = 0.00122 g/ cm^3 (Microshield Default) — Reference 8, page 5-8
7. The Dose Contribution Factors (DCFs) for Iodine isotopes 131 – 135 reproduced in Table 1 are taken from Reference 2, Table 2.1 (pg. 136) Inhalation - Effective Committed Dose Equivalent.

Table 1: Iodine Dose Contribution Factors

Isotope	Effective CDE (Sv/Bq)
I-131	8.89E-09
I-132	1.03E-10
I-133	1.58E-09
I-134	3.55E-11
I-135	3.32E-10

8. The source nuclide inventory is taken from Reference 3, Tables 5-10 and 5-15 as adjusted depending on the case assumptions. See the Method section for a detailed source description.

References

1. NEI 99-01 Rev. 6, "Development of Emergency Action Levels", November 2012.

2. Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion." 1988.

3. 0450-0100004WN, Radiation Analysis Manual, Rev. 5, May 1997, STI: 30521679.

4. TS 3/4.4.8, Reactor Coolant System Specific Activity.

5. NC-7007, Rev. 10, "MSLB Containment Pressure and Temperature Analysis", STI: 32839688.

6. Drawing 6C189N05006, Rev. 9, "Gen. Arrangement Reactor Containment Building Plan at EL. 68'-0" Area G"; STI: 292218.

7. MicroShield 9.07 SQA Package, April 22, 2015, STI: 34116064.

8. MicroShield User's Manual; distributed with the software.

Assumptions

1. The modeled release is an instantaneous flash of the assumed RCS nuclide inventory to a uniform airborne distribution within an empty containment vessel. No shielding is modeled within the containment volume other than air at standard atmospheric conditions. The model containment volume is approximately the containment free volume (see Assumption 2).

2. It is assumed that the containment height is 183.9 ft, which is a "hydraulic" height from Reference 5. This will make the containment volume slightly smaller than the value reported in Reference 5 for containment free volume, which slightly increases the source term concentration.

3. Containment radiation monitor instrument sensitivity and uncertainty are beyond the scope of this calculation. It is assumed that the instruments will accurately read the radiation field.

4. Aerosol behavior and other chemical phenomena which may impact the axial distribution of various nuclides within containment are not modeled.

5. Noble gas concentration for 1% failed fuel from Reference 3, Table 5-15 are assumed to be the TS allowable concentrations. This is justified further in the Calculation section for Case 1.

6. The dose point is assumed to be at the radial edge and axial mid-plane of the problem. The dose point height has a negligible impact on the results away from the edges, as shown in the sensitivity cases for the Case 3 output (pg. B-10). The dose point is removed radially from the containment volume by a nominal 1 cm to avoid potential issues with placing a dose point on a material interface.

7. The actual placement of monitors RE-8050 and RE-8051 is approximately 2' 6" inward from the containment wall. Immersion and reflector effects on dose rates are assumed to be within the source term uncertainty for each scenario. MS9 has a limited ability to calculate an immersion dose.

Method of Analysis

This calculation does not contain a change in the methodology as described in the UFSAR. This calculation produces expected containment radiation monitor responses for 3 postulated scenarios. The scenarios are defined based on the guidance in Reference 1. One Microshield case is run for each scenario assuming an instantaneous release of the RCS contents into the containment atmosphere per Assumption 1. The containment atmosphere is modeled as a right circular cylinder with the same radius as the inner radius of containment and a height conservatively representative of the containment free volume per Assumption 2. One dose point is placed according to Assumption 6. Each case represents indication by the containment radiation monitors of a loss of fission product barrier.

Case 1: Loss of RCS Barrier Only – PWR RCS Barrier Threshold Loss 3.A
Case 2: Loss of RCS and Fuel Cladding Barrier – PWR Fuel Clad Barrier Threshold Loss 3.A
Case 3: Challenged Containment Barrier – PWR Containment Barrier Threshold Potential Loss 3.A

The sources for the different cases are described below.

Case 1: RCS Water with Maximum TS Allowable Specific Activity

For the TS concentration release case, the Iodine activities are determined based on a Dose Equivalent Iodine (DEI-131) of 1 μCi/gm per TS 3/4.4.8 (Reference 4) and the remaining activities are determined based on the noble gas concentrations listed in Table 5-15 of Reference 3. TS 3/4.4.8 has two requirements for RCS activity. The limit in 3/4.4.8.a is that the DEI-131 is less than 1 μCi/gm, and 3/4.4.8.b requires that the total specific activity is less than 100 / $\bar{E}$ where $\bar{E}$ is the average energy per disintegration in MeV / Disintegration (Dis).

According to Reference 3 page 5-26, the value of $\bar{E}$ for Table 5-15 is approximately 0.36 MeV/ Dis. The resulting value of 100/ $\bar{E}$ is 278 μCi/gm which is slightly less than the total specific activity of Table 5-15. Since the Table 5-15 values produce a DEI-131 greater than 1, the DEI-131 limit is treated as the bounding TS limit for this case. Noble gas nuclides which are not included in the DEI-131 calculation remain at the Table 5-15 concentrations for the TS limit case. Iodine isotopes 131-135 are uniformly reduced from the Table 5-15 values to a DEI-131 level of 1 μCi/gm.

Case 2: RCS Water with a DEI-131 of 300 μCi/gm and 2% Total Gap Release.

This case is intended to indicate that a significant amount of clad damage has occurred. Guidance in Reference 1, page 105 states that the DEI-131 of 300 μCi/gm is intended to represent a 2% to 5% fuel clad damage. The DEI-131 for a 2% Total Gap Release (Reference 3, Table 5-10 represents 100%) is approximately 1202 μCi/gm. Therefore the five main Iodine isotopes are scaled down to a DEI-131 value of 300 μCi/gm. A 2% total gap release is assumed for the remaining nuclides.

Case 3: RCS Water with a 20% Total Gap Release.

This case is intended to represent the potential for a major release of radioactivity. Reference 1 suggests a noble gas and Iodine activity associated with 20% of the fuel clad failure in the containment. Table 5-10 inventories are simply multiplied by a factor of 0.2 for this case.

Equations:

DEI-131 is calculated by determining the ratios of each Iodine isotope's DCF to that of I-131 as shown in Equation 1.

$$\text{Equation 1: } R_i = \frac{DCF_i}{DCF_{131}}$$

The Ratios are then used to determine DEI-131 according to Equation 2 where "A" is specific activity in units of µCi/gm.

$$\text{Equation 2: } DEI\text{-}131 = A_{131} + A_{132}R_{132} + A_{133}R_{133} + A_{134}R_{134} + A_{135}R_{135}$$

Adjusting Iodine specific activities from a known DEI-131 level to the level desired for a calculation is done using Equation 3 where A_N is the new activity and A_I is the initial activity.

$$\text{Equation 3: } A_N = A_I \frac{DEI-131\_N}{DEI-131\_I}$$

Software:

Microshield Version 9.07 (MS9) (Reference 7) is used to calculate the exposure rate in mR/hr at the dose point. MS9 is executed on a Windows 7 PC. The software output is included as Appendix B. The software calculates and bins the source photons by energy group based on the input nuclide inventory. The software provides a multi-group point-kernel solution for the problem with and without the buildup factor (the increase in measured radiation dose due the contribution of scattered particles). The solutions with buildup are reported as the results of this calculation.

Calculation

Dimensions:
The MS9 model is a right circular cylinder defined by the height and radius specified in inputs 1 and 2 respectively.

Dose Point Location:
The dose point height is half of the containment height and just outside the containment radius per Assumption 6.

DP_Y = 183.9 ft / 2 = 91.95 ft. (2802.6 cm)
DP_X = 2286.0 cm + 1.0 cm = 2287.0 cm.
DP_Z = 0.0 cm

Buildup and Quadrature:
Buildup is calculated in the source region which is the containment volume since there is no shield in this problem. The problem quadrature (spatial mesh cells) is set at 20, 30, and 30 for the radial, circumferential, and axial directions respectively. Sensitivity cases were executed at 5 quadrature values for each direction to confirm that increased meshing from the selected mesh size did not affect the solution. More details on proper model mesh sizing are contained in Reference 8.

DEI-131 Ratios:
Ratios calculated for the purposes of DEI-131 calculation according to Equation 1 are reported in Table 2 below:

Table 2: DEI-131 Ratios

Isotope	DEI-131 Ratios
I-132	1.159E-02
I-133	1.777E-01
I-134	3.993E-03
I-135	3.735E-02

Case 1:
The following calculations justify that 1% failed fuel (Reference 3, Table 5-15) values are acceptable for the activity of noble gases for the TS Limit case.
TS Specific Activity Limit for gross specific activity: TS 3/4.4.8.b limit = 100 / Ē

TS 3/4.4.8.b limit = 100 / 0.36 = 278 μCi/gm

The sum of all nuclides in Table 5-15 of Reference 3 would be slightly greater than this limit. However, reducing the DEI-131 to 1 μCi/gm and limiting the population to Iodine and noble gas results in a gross specific activity of 275 μCi/gm. Therefore, these values are an acceptable assumption for TS allowable concentrations. This is demonstrated on Table 3. This calculation does not attempt to calculate activities at the "100 / Ē" limit because there are multiple concentrations that could meet this limit and any change in concentration also changes Ē.

Table 3 below lists the specific activity for each Iodine and noble gas nuclide with 1% failed fuel during normal operations with the CVCS in service in column B. This is different from a total gap activity release associated with an accident. The DEI-131 and total activity are also calculated and presented at the bottom.

Adjusted Iodine concentrations for a DEI-131 of 1 μCi/gm displayed in column C are calculated by dividing the 1% failed fuel values by the original DEI-131 from Reference 3 (calculated near the bottom of column B). The total RCS activity in column D is calculated by multiplying the Adjusted Activity with the RCS mass of 2.65E+8 grams. These are the activity values used in the MS9 model for Case 1. The external source file for Case 1 is *AST_DEI_1.mxd*. The case file is *AST_1_Cont.msd*.

Table 3: Iodine and Noble Gas TS Allowable Activities

A	B	C	D
Nuclide	1% Failed Fuel Activity (μCi/gm)	Adjusted DEI-131 Activity (μCi/gm)	Total RCS Activity (Ci)
I-131	1.7	0.677	1.79E+02
I-132	2.4	0.9558	2.53E+02
I-133	2.8	1.1151	2.96E+02
I-134	0.52	0.2071	5.49E+01
I-135	7.6	3.0267	8.02E+02
Kr83m	0.37	0.37	9.81E+01
Kr85	7.6	7.6	2.01E+03
Kr85m	1.5	1.5	3.98E+02
Kr87	0.98	0.98	2.60E+02
Kr88	2.8	2.8	7.42E+02
Kr89	0.084	0.084	2.23E+01
Xe131m	2.8	2.8	7.42E+02
Xe133	240	240	6.36E+04
Xe133m	4.2	4.2	1.11E+03
Xe135	7.6	7.6	2.01E+03
Xe135m	0.4	0.4	1.06E+02
Xe137	0.16	0.16	4.24E+01
Xe138	0.58	0.58	1.54E+02
DEI-131 (Eqn. 2)	2.511	1.000	
Total	284	275	
	Ref. 3 Values	(For Iodine Only) B ÷ 2.511	$C \times \left(\dfrac{2.65x10^8\, gm}{1x10^3\, \mu Ci/Ci} \right)$

Case 2:
Case 2 will use 2% of the total activities listed in Reference 3, Table 5-10, but adjust the DEI-131 lower to a value of 300 μCi/gm. The transition from Table 5-15 to Table 5-10 from Reference 3 is due the postulated loss of the cladding fission product barrier specified for this case. Table 4 on the following page documents the nuclide inventory for Cases 2 and 3. Column B is a reproduction of the inventory from Reference 3, Table 5-10. Column C is 2% of that inventory which results in a DEI-131 calculation of 1202 μCi/gm.

Column D shows the adjusted Iodine inventories for a DEI-131 of 300 μCi/gm. This is calculated according to Equation 3. Iodine-130 is excluded from the adjustment as it is not typically included in DEI-131 calculations and has a concentration much smaller than the other five isotopes. Also, note that due to the units in Table 4, calculation of the DEI-131 requires division by the RCS mass and scaling from Ci to μCi. The external source file for Case 2 is *Total_Gap_2.mxd*. The case file is *AST_2_Cont.msd*.

Case 3:
Case 3 will use 20% of the total activities listed in Column 2 of Table 4. These values are reported in Column 5. The external source file for Case 3 is *Total_Gap_20.mxd*. The case file is *AST_20_Cont.msd*.

Table 4: Cases 2 and 3 Activities

A	B	C	D	E
Nuclide	Total Gap 100% Release (Ci)	2% Gap Release (Ci)	Case 2: 2% Gap Release Adjusted DEI (Ci)	Case 3: 20% Gap Release (Ci)
I-130	4.30E+05	8.60E+03	8.60E+03	8.60E+04
I-131	1.10E+07	2.20E+05	5.49E+04	2.20E+06
I-132	1.50E+07	3.00E+05	7.49E+04	3.00E+06
I-133	2.20E+07	4.40E+05	1.10E+05	4.40E+06
I-134	2.40E+07	4.80E+05	1.20E+05	4.80E+06
I-135	2.00E+07	4.00E+05	9.98E+04	4.00E+06
Kr83m	1.30E+06	2.60E+04	2.60E+04	2.60E+05
Kr85	3.70E+05	7.40E+03	7.40E+03	7.40E+04
Kr85m	2.90E+06	5.80E+04	5.80E+04	5.80E+05
Kr87	5.50E+06	1.10E+05	1.10E+05	1.10E+06
Kr88	7.80E+06	1.56E+05	1.56E+05	1.56E+06
Kr89	9.50E+06	1.90E+05	1.90E+05	1.90E+06
Xe131m	1.10E+05	2.20E+03	2.20E+03	2.20E+04
Xe133	2.20E+07	4.40E+05	4.40E+05	4.40E+06
Xe133m	6.80E+05	1.36E+04	1.36E+04	1.36E+05
Xe135	5.50E+06	1.10E+05	1.10E+05	1.10E+06
Xe135m	4.20E+06	8.40E+04	8.40E+04	8.40E+05
Xe137	1.90E+07	3.80E+05	3.80E+05	3.80E+06
Xe138	1.80E+07	3.60E+05	3.60E+05	3.60E+06
DEI-131 (μCi/gm)		1202	300	12020
	Ref. 3 Values	B x 0.02	(For Iodine 131-135 Only) $C \times \left(\frac{300}{1202}\right)$	B x 0.20

Conclusion

Cases were executed for each of the 3 source terms. See Appendix B for the detailed Microshield output. The results with buildup are reported below. They are rounded to the nearest R/hr.

Case 1 Exposure Rate: **13 R/hr**

Case 2 Exposure Rate: **2144 R/hr**

Case 3 Exposure Rate: **45,040 R/hr**

	0PGP04-ZA-0307	Rev. 6	
Preparation of Calculations			
Form 3	Calculation Impact Assessment		Page 1 of 1

Calculation No.: 15-RA-011 Rev.: 0 Page: 14

Inter-discipline Coordination Required? ☐ YES ☒ NO

	Initial/Date		Initial/Date
☐ Risk and Reliability Analysis		☐ Thermal Hydraulics (Ch 15 Safety Analysis, HELBA)	
☐ Reactor Engineering (including Radiological Impact)		☐ Civil Engineering	
☐ Mechanical Engineering		☐ Electrical Engineering	
☐ Equipment Qualification		☐ I&C Engineering	
		☐ Other:	

List sections reviewed of applicable documents/applicable attributes of the calculation/CRs generated:
This calculation was prepared in support of a Emergency Plan revision, so impacts are anticipated. The entire Emergency Plan is being updated to the guidance in Reference 1 (NEI 99-01). See CR 13-10171.

TECHNICAL SPECIFICATIONS SECTIONS REVIEWED/IMPACT:
None

UFSAR SECTIONS REVIEWED/IMPACT:
Sections 3.12, 12.3 and 12.4 / No Impact. Results of this calculation are within the range of the subject detectors.

SER SECTIONS REVIEWED/IMPACT:
None

DESIGN BASIS DOCUMENTS REVIEWED/IMPACT:
5Z149ZB01000, "Radiation Monitoring System". Potential impact to stated alarm setpoint (4C.15.2.4 & 4C.16.2.4)

PROCEDURES REVIEWED/IMPACT:
0ERP01-ZV-IN01 / EAL-5, and various Initiating Conditions.

OTHER CALCULATIONS REVIEWED/IMPACT:
91-RA-001, 03-ZE-003, 12-RA-003 / Results for RT-8050 & RT-8051 from 91-RA-001 are superseded. 03-ZE-003, Action 13-10171-67 to incorporate new reference source.

OTHER DOCUMENTS REVIEWED/IMPACT:
Emergency Plan impacts are anticipated and will be implemented by action 13-10171-66.

certrec.com FILES SEARCHED AND SEARCH PARAMETERS USED:
None

	0PGP04-ZA-0307	Rev. 6
Preparation of Calculations		
Form 4	Checkers Comments	Page 1 of 1

	Calculation No.:	15-RA-011	Rev.:	0	Page:	15

The Following attributes are required by this procedure

Yes	N/A	
☒		The calculation has been assigned a Priority Code (3.1.2)
☒		The calculation has been assigned a Quality Class (3.1.3)
☒		The calculation contains the pages identified in (3.1.4)
☒		The calculation contains an Objective and Scope (3.1.5.1)
☒		The calculation contains a Summary of Results (3.1.5.2)
☒		The calculation describes the Method of Analysis (3.1.5.3)
☒		The calculation correctly identifies inputs (3.1.5.4)
☐	☒	A CR action has been issued to track inputs that require confirmation (3.1.5.4)
☒		The calculation identifies the references (3.1.5.5)
☒		The calculation clearly identifies assumptions (3.1.5.6)
☐	☒	A CR action has been issued to track assumptions that require confirmation (3.1.5.6)
☒		The calculation adequately presents the results (3.1.5.7)
☐	☒	Computer analysis meets the requirements of 3.1.5.8
☒		The calculation contains an impact review (3.1.5.9)
☒		The Originator and Checker are qualified to Engineering Support Certification 9569, "Design and Engineering Calculations" as shown in the Qual King database.

Checker's calculation summary

The calculation makes reasonable assumptions about source term and containment dispersion. All of the inputs are documented, and the calculations performed are correct.

Appendix A: CD File Listing

Total Number of Pages in Attachment: 1

Table A-1: CD File Listing

File Name	Description	MD5 File Hash
AST_1_Cont.msd	Case file for RCS water with TS allowable specific activity (Case 1)*	44036F88AA75F3B4B482889C5BB90B35
AST_2_Cont.msd	Case file for RCS water with 2% Total Fuel Rod Gap Release and DEI-131 of 300 (Case 2)*	B6FF6113D416F2FA7FD1A6FF1DBA37AF
AST_20_Cont.msd	Case file for RCS water with 20% Total Fuel Rod Gap Release. (Case 3)*	8D0DB3266E3837ADC6BDE5824727ACFB
AST_DEI_1.mxd	External Source file for RCS water with TS allowable specific activity (Case 1)	D81CC3FB692A971DBE4D57370EE12ACF
Total_Gap_2.mxd	External Source file for RCS water with 2% Total Fuel Rod Gap Release and DEI-131 of 300 (Case 2)	142E3F0BBA6125E6586BF230D4161559
Total_Gap_20.mxd	External Source file for RCS water with 20% Total Fuel Rod Gap Release. (Case 3)	F7D4B9E4EF5257D4037D1FC7732D14F6

***NOTE:** Output is included in the ".msd" files and is accessed via the MS9 interface.

The directory listing for the attached CD is below:

```
Volume in drive D is 15-RA-011_R0
 Volume Serial Number is 2058-D63B

 Directory of D:\

04/28/2015  03:04 PM              3,721 AST_1_Cont.msd
04/28/2015  03:00 PM              7,212 AST_20_Cont.msd
04/28/2015  03:03 PM              3,832 AST_2_Cont.msd
04/28/2015  01:01 PM              2,059 AST_DEI_1.mxd
04/28/2015  11:31 AM              2,079 Total_Gap_2.mxd
04/28/2015  09:06 AM              2,080 Total_Gap_20.mxd
               6 File(s)         20,983 bytes
               0 Dir(s)               0 bytes free
```

Appendix B: Microshield Output

Total Number of Pages in Attachment: 10

MicroShield 9.07
STPNOC (9.07-0000)

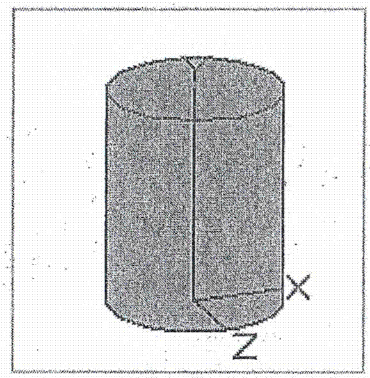
Date	By	Checked

Filename	Run Date	Run Time	Duration
AST_1_Cont.msd	April 28, 2015	3:03:43 PM	00:00:01

Project Info	
Case Title	1 NJH 4-29-15 Cont. Rad Response
Description	Case ~~2~~ - Cont. Volume w/ DEI = 1 & Table 5-15 noble gases
Geometry	7 - Cylinder Volume - Side Shields

Source Dimensions	
Height	5.6e+3 cm (183 ft 10.8 in)
Radius	2.3e+3 cm (75 ft)

Dose Points			
A	X	Y	Z
#1	2.3e+3 cm (75 ft 0.4 in)	2.8e+3 cm (91 ft 11.4 in)	0.0 cm (0 in)



Shields			
Shield N	Dimension	Material	Density
Source	9.20e+10 cm³	Air	0.00122
Transition		Air	0.00122
Air Gap		Air	0.00122

Source Input: Grouping Method - Standard Indices
Number of Groups: 25
Lower Energy Cutoff: 0.015
Photons < 0.015: Included
Library: Grove

Nuclide	Ci	Bq	μCi/cm³	Bq/cm³
I-131	1.7900e+002	6.6230e+012	1.9452e-003	7.1971e+001
I-132	2.5300e+002	9.3610e+012	2.7493e-003	1.0172e+002
I-133	2.9600e+002	1.0952e+013	3.2166e-003	1.1901e+002
I-134	5.4900e+001	2.0313e+012	5.9659e-004	2.2074e+001
I-135	8.0200e+002	2.9674e+013	8.7152e-003	3.2246e+002
Kr-83m	9.8100e+001	3.6297e+012	1.0660e-003	3.9443e+001
Kr-85	2.0100e+003	7.4370e+013	2.1842e-002	8.0816e+002
Kr-85m	3.9800e+002	1.4726e+013	4.3250e-003	1.6002e+002
Kr-87	2.6000e+002	9.6200e+012	2.8254e-003	1.0454e+002
Kr-88	7.4200e+002	2.7454e+013	8.0632e-003	2.9834e+002
Kr-89	2.2300e+001	8.2510e+011	2.4233e-004	8.9662e+000
Xe-131m	7.4200e+002	2.7454e+013	8.0632e-003	2.9834e+002
Xe-133	6.3600e+004	2.3532e+015	6.9113e-001	2.5572e+004
Xe-133m	1.1100e+003	4.1070e+013	1.2062e-002	4.4630e+002
Xe-135	2.0100e+003	7.4370e+013	2.1842e-002	8.0816e+002
Xe-135m	1.0600e+002	3.9220e+012	1.1519e-003	4.2620e+001
Xe-137	4.2400e+001	1.5688e+012	4.6075e-004	1.7048e+001
Xe-138	1.5400e+002	5.6980e+012	1.6735e-003	6.1919e+001

Buildup: The material reference is Source
Integration Parameters

Radial	20
Circumferential	30

15-RA-011: Appendix B

Y Direction (axial)	30

Results

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.015	1.551e+14	6.254e+03	7.137e+03	5.364e+02	6.122e+02	4.683e+02	5.344e+02	4.683e+00	5.344e+00
0.03	1.172e+15	3.135e+05	5.326e+05	3.107e+03	5.279e+03	2.713e+03	4.608e+03	2.713e+01	4.608e+01
0.08	8.638e+14	8.212e+05	1.541e+06	1.300e+03	2.439e+03	1.135e+03	2.129e+03	1.135e+01	2.129e+01
0.1	5.414e+10	6.566e+01	1.136e+02	1.004e-01	1.738e-01	8.769e-02	1.517e-01	8.769e-04	1.517e-03
0.15	1.319e+13	2.479e+04	3.851e+04	4.083e+01	6.342e+01	3.565e+01	5.536e+01	3.565e-01	5.536e-01
0.2	8.103e+13	2.077e+05	2.904e+05	3.666e+02	5.126e+02	3.201e+02	4.475e+02	3.201e+00	4.475e+00
0.3	5.778e+12	2.295e+04	2.958e+04	4.353e+01	5.611e+01	3.801e+01	4.898e+01	3.801e-01	4.898e-01
0.4	1.520e+13	8.234e+04	1.012e+05	1.604e+02	1.972e+02	1.401e+02	1.721e+02	1.401e+00	1.721e+00
0.5	1.874e+13	1.290e+05	1.537e+05	2.533e+02	3.016e+02	2.211e+02	2.633e+02	2.211e+00	2.633e+00
0.6	1.629e+13	1.365e+05	1.590e+05	2.664e+02	3.104e+02	2.325e+02	2.710e+02	2.325e+00	2.710e+00
0.8	2.066e+13	2.355e+05	2.663e+05	4.479e+02	5.065e+02	3.910e+02	4.421e+02	3.910e+00	4.421e+00
1.0	1.893e+13	2.737e+05	3.041e+05	5.045e+02	5.605e+02	4.404e+02	4.894e+02	4.404e+00	4.894e+00
1.5	2.357e+13	5.241e+05	5.669e+05	8.818e+02	9.537e+02	7.698e+02	8.326e+02	7.698e+00	8.326e+00
2.0	2.324e+13	6.997e+05	7.446e+05	1.082e+03	1.151e+03	9.447e+02	1.005e+03	9.447e+00	1.005e+01
3.0	1.631e+12	7.506e+04	7.856e+04	1.018e+02	1.066e+02	8.890e+01	9.305e+01	8.890e-01	9.305e-01
4.0	3.695e+10	2.292e+03	2.377e+03	2.835e+00	2.940e+00	2.475e+00	2.567e+00	2.475e-02	2.567e-02
Totals	**2.430e+15**	**3.555e+06**	**4.816e+06**	**9.095e+03**	**1.305e+04**	**7.940e+03**	**1.140e+04**	**7.940e+01**	**1.140e+02**

MicroShield 9.07
STPNOC (9.07-0000)

Date	By	Checked

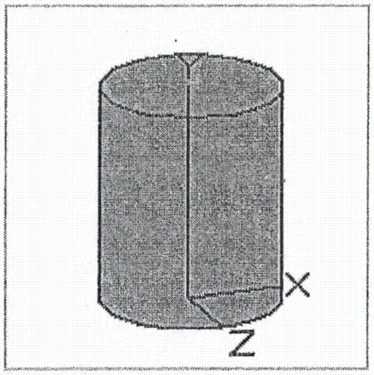
Filename	Run Date	Run Time	Duration
AST_2_Cont.msd	April 28, 2015	3:01:50 PM	00:00:01

Project Info	
Case Title	Cont. Rad Response
Description	Case 2 - Cont. Volume with 2% Failed Fuel & RCS, DEI = 300
Geometry	7 - Cylinder Volume - Side Shields

Source Dimensions	
Height	5.6e+3 cm (183 ft 10.8 in)
Radius	2.3e+3 cm (75 ft)

Dose Points			
A	X	Y	Z
#1	2.3e+3 cm (75 ft 0.4 in)	2.8e+3 cm (91 ft 11.4 in)	0.0 cm (0 in)

Shields			
Shield N	Dimension	Material	Density
Source	9.20e+10 cm³	Air	0.00122
Transition		Air	0.00122
Air Gap		Air	0.00122



Source Input: Grouping Method - Standard Indices
Number of Groups: 25
Lower Energy Cutoff: 0.015
Photons < 0.015: Included
Library: Grove

Nuclide	Ci	Bq	μCi/cm³	Bq/cm³
I-130	8.6000e+003	3.1820e+014	9.3454e-002	3.4578e+003
I-131	5.4900e+004	2.0313e+015	5.9659e-001	2.2074e+004
I-132	7.4900e+004	2.7713e+015	8.1392e-001	3.0115e+004
I-133	1.1000e+005	4.0700e+015	1.1953e+000	4.4228e+004
I-134	1.2000e+005	4.4400e+015	1.3040e+000	4.8249e+004
I-135	9.9800e+004	3.6926e+015	1.0845e+000	4.0127e+004
Kr-83m	2.6000e+004	9.6200e+014	2.8254e-001	1.0454e+004
Kr-85	7.4000e+003	2.7380e+014	8.0414e-002	2.9753e+003
Kr-85m	5.8000e+004	2.1460e+015	6.3027e-001	2.3320e+004
Kr-87	1.1000e+005	4.0700e+015	1.1953e+000	4.4228e+004
Kr-88	1.5600e+005	5.7720e+015	1.6952e+000	6.2723e+004
Kr-89	1.9000e+005	7.0300e+015	2.0647e+000	7.6394e+004
Xe-131m	2.2000e+003	8.1400e+013	2.3907e-002	8.8456e+002
Xe-133	4.4000e+005	1.6280e+016	4.7814e+000	1.7691e+005
Xe-133m	1.3600e+004	5.0320e+014	1.4779e-001	5.4682e+003
Xe-135	1.1000e+005	4.0700e+015	1.1953e+000	4.4228e+004
Xe-135m	8.4000e+004	3.1080e+015	9.1281e-001	3.3774e+004
Xe-137	3.8000e+005	1.4060e+016	4.1294e+000	1.5279e+005
Xe-138	3.6000e+005	1.3320e+016	3.9120e+000	1.4475e+005

Buildup: The material reference is Source
Integration Parameters

Radial	20

Circumferential	30
Y Direction (axial)	30

Results

Energy (MeV)	**Activity (Photons/sec)**	**Fluence Rate MeV/cm²/sec No Buildup**	**Fluence Rate MeV/cm²/sec With Buildup**	**Exposure Rate mR/hr No Buildup**	**Exposure Rate mR/hr With Buildup**	**Absorbed Dose Rate mrad/hr No Buildup**	**Absorbed Dose Rate mrad/hr With Buildup**	**Absorbed Dose Rate mGy/hr No Buildup**	**Absorbed Dose Rate mGy/hr With Buildup**
0.015	2.953e+15	1.191e+05	1.359e+05	1.022e+04	1.166e+04	8.918e+03	1.018e+04	8.918e+01	1.018e+02
0.03	9.625e+15	2.574e+06	4.373e+06	2.551e+04	4.334e+04	2.227e+04	3.783e+04	2.227e+02	3.783e+02
0.08	6.028e+15	5.731e+06	1.075e+07	9.069e+03	1.702e+04	7.917e+03	1.486e+04	7.917e+01	1.486e+02
0.1	1.138e+13	1.380e+04	2.388e+04	2.112e+01	3.653e+01	1.844e+01	3.189e+01	1.844e-01	3.189e-01
0.15	2.828e+15	5.314e+06	8.253e+06	8.750e+03	1.359e+04	7.639e+03	1.186e+04	7.639e+01	1.186e+02
0.2	7.473e+15	1.916e+07	2.679e+07	3.381e+04	4.728e+04	2.952e+04	4.127e+04	2.952e+02	4.127e+02
0.3	5.138e+15	2.041e+07	2.630e+07	3.871e+04	4.989e+04	3.379e+04	4.356e+04	3.379e+02	4.356e+02
0.4	9.450e+15	5.118e+07	6.290e+07	9.973e+04	1.226e+05	8.706e+04	1.070e+05	8.706e+02	1.070e+03
0.5	1.306e+16	8.995e+07	1.071e+08	1.766e+05	2.103e+05	1.541e+05	1.836e+05	1.541e+03	1.836e+03
0.6	7.698e+15	6.447e+07	7.514e+07	1.258e+05	1.467e+05	1.099e+05	1.280e+05	1.099e+03	1.280e+03
0.8	1.390e+16	1.584e+08	1.792e+08	3.013e+05	3.408e+05	2.631e+05	2.975e+05	2.631e+03	2.975e+03
1.0	7.032e+15	1.017e+08	1.130e+08	1.874e+05	2.082e+05	1.636e+05	1.818e+05	1.636e+03	1.818e+03
1.5	6.708e+15	1.492e+08	1.613e+08	2.510e+05	2.715e+05	2.191e+05	2.370e+05	2.191e+03	2.370e+03
2.0	1.120e+16	3.373e+08	3.589e+08	5.217e+05	5.551e+05	4.554e+05	4.846e+05	4.554e+03	4.846e+03
3.0	1.244e+15	5.727e+07	5.994e+07	7.770e+04	8.133e+04	6.783e+04	7.100e+04	6.783e+02	7.100e+02
4.0	3.148e+14	1.953e+07	2.025e+07	2.416e+04	2.505e+04	2.109e+04	2.187e+04	2.109e+02	2.187e+02
Totals	**1.047e+17**	**1.082e+09**	**1.214e+09**	**1.891e+06**	**2.144e+06**	**1.651e+06**	**1.872e+06**	**1.651e+04**	**1.872e+04**

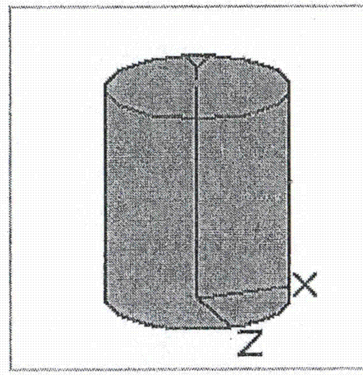
MicroShield 9.07
STPNOC (9.07-0000)

Date	By	Checked

Filename	Run Date	Run Time	Duration
AST_20_Cont.msd	April 28, 2015	2:58:12 PM	00:00:04

Project Info	
Case Title	Cont. Rad Response
Description	Case 3 - Cont. Volume with 20% Failed Fuel & Failed RCS
Geometry	7 - Cylinder Volume - Side Shields

Source Dimensions	
Height	5.6e+3 cm (183 ft 10.8 in)
Radius	2.3e+3 cm (75 ft)



Dose Points			
A	X	Y	Z
#1	2.3e+3 cm (75 ft 0.4 in)	2.8e+3 cm (91 ft 11.4 in)	0.0 cm (0 in)

Shields			
Shield N	Dimension	Material	Density
Source	9.20e+10 cm³	Air	0.00122
Transition		Air	0.00122
Air Gap		Air	0.00122

Source Input: Grouping Method - Standard Indices
Number of Groups: 25
Lower Energy Cutoff: 0.015
Photons < 0.015: Included
Library: Grove

Nuclide	Ci	Bq	μCi/cm³	Bq/cm³
I-130	8.6000e+004	3.1820e+015	9.3454e-001	3.4578e+004
I-131	2.2000e+006	8.1400e+016	2.3907e+001	8.8456e+005
I-132	3.0000e+006	1.1100e+017	3.2600e+001	1.2062e+006
I-133	4.4000e+006	1.6280e+017	4.7814e+001	1.7691e+006
I-134	4.8000e+006	1.7760e+017	5.2161e+001	1.9299e+006
I-135	4.0000e+006	1.4800e+017	4.3467e+001	1.6083e+006
Kr-83m	2.6000e+005	9.6200e+015	2.8254e+000	1.0454e+005
Kr-85	7.4000e+004	2.7380e+015	8.0414e-001	2.9753e+004
Kr-85m	5.8000e+005	2.1460e+016	6.3027e+000	2.3320e+005
Kr-87	1.1000e+006	4.0700e+016	1.1953e+001	4.4228e+005
Kr-88	1.5600e+006	5.7720e+016	1.6952e+001	6.2723e+005
Kr-89	1.9000e+006	7.0300e+016	2.0647e+001	7.6394e+005
Xe-131m	2.2000e+004	8.1400e+014	2.3907e-001	8.8456e+003
Xe-133	4.4000e+006	1.6280e+017	4.7814e+001	1.7691e+006
Xe-133m	1.3600e+005	5.0320e+015	1.4779e+000	5.4682e+004
Xe-135	1.1000e+006	4.0700e+016	1.1953e+001	4.4228e+005
Xe-135m	8.4000e+005	3.1080e+016	9.1281e+000	3.3774e+005
Xe-137	3.8000e+006	1.4060e+017	4.1294e+001	1.5279e+006
Xe-138	3.6000e+006	1.3320e+017	3.9120e+001	1.4475e+006

Buildup: The material reference is Source
Integration Parameters

Radial	20

Circumferential	30
Y Direction (axial)	30

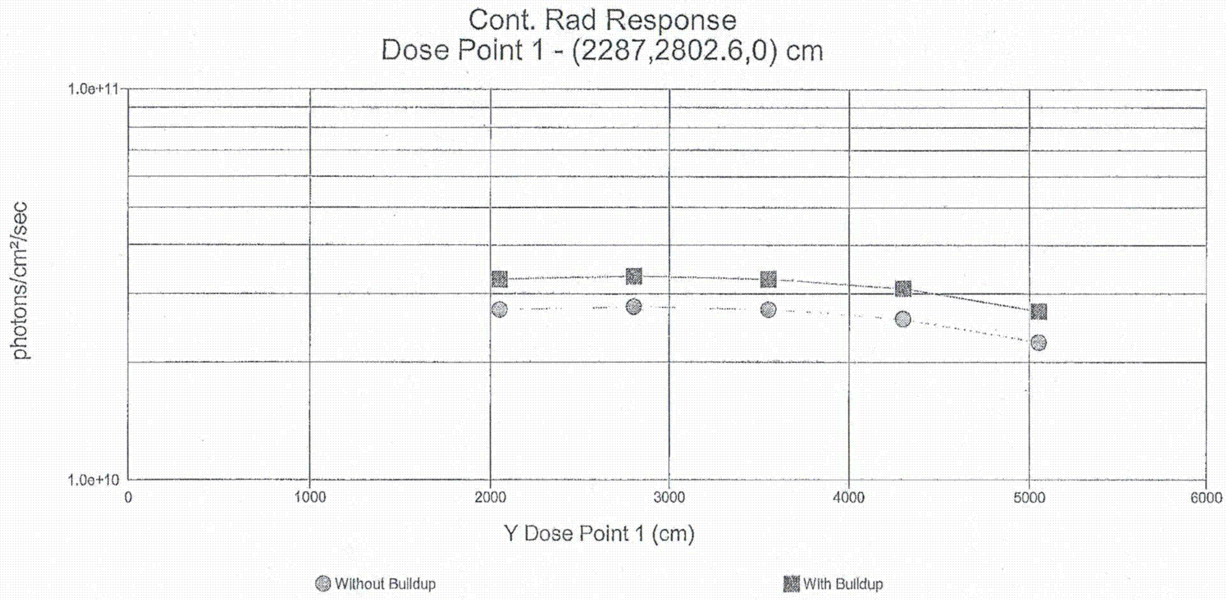
Results

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.015	3.024e+16	1.219e+06	1.392e+06	1.046e+05	1.194e+05	9.131e+04	1.042e+05	9.131e+02	1.042e+03
0.03	1.026e+17	2.743e+07	4.660e+07	2.719e+05	4.619e+05	2.373e+05	4.032e+05	2.373e+03	4.032e+03
0.08	6.188e+16	5.883e+07	1.104e+08	9.310e+04	1.747e+05	8.127e+04	1.525e+05	8.127e+02	1.525e+03
0.1	1.138e+14	1.380e+05	2.388e+05	2.112e+02	3.653e+02	1.844e+02	3.189e+02	1.844e+00	3.189e+00
0.15	3.488e+16	6.555e+07	1.018e+08	1.079e+05	1.676e+05	9.423e+04	1.464e+05	9.423e+02	1.464e+03
0.2	8.110e+16	2.079e+08	2.907e+08	3.670e+05	5.131e+05	3.204e+05	4.479e+05	3.204e+03	4.479e+03
0.3	6.298e+16	2.501e+08	3.224e+08	4.745e+05	6.116e+05	4.142e+05	5.339e+05	4.142e+03	5.339e+03
0.4	1.692e+17	9.166e+08	1.126e+09	1.786e+06	2.195e+06	1.559e+06	1.916e+06	1.559e+04	1.916e+04
0.5	2.849e+17	1.962e+09	2.336e+09	3.850e+06	4.585e+06	3.361e+06	4.003e+06	3.361e+04	4.003e+04
0.6	2.353e+17	1.971e+09	2.297e+09	3.847e+06	4.484e+06	3.359e+06	3.914e+06	3.359e+04	3.914e+04
0.8	4.715e+17	5.373e+09	6.076e+09	1.022e+07	1.156e+07	8.922e+06	1.009e+07	8.922e+04	1.009e+05
1.0	1.950e+17	2.820e+09	3.133e+09	5.198e+06	5.775e+06	4.538e+06	5.042e+06	4.538e+04	5.042e+04
1.5	1.610e+17	3.580e+09	3.872e+09	6.023e+06	6.515e+06	5.258e+06	5.687e+06	5.258e+04	5.687e+04
2.0	1.376e+17	4.145e+09	4.411e+09	6.410e+06	6.821e+06	5.596e+06	5.955e+06	5.596e+04	5.955e+04
3.0	1.244e+16	5.727e+08	5.994e+08	7.770e+05	8.133e+05	6.783e+05	7.100e+05	6.783e+03	7.100e+03
4.0	3.148e+15	1.953e+08	2.025e+08	2.416e+05	2.505e+05	2.109e+05	2.187e+05	2.109e+03	2.187e+03
Totals	**2.044e+18**	**2.215e+10**	**2.493e+10**	**3.977e+07**	**4.504e+07**	**3.472e+07**	**3.932e+07**	**3.472e+05**	**3.932e+05**
	Sensitivity	**Variable**	**Y Dose Point 1**	**(1 of 5)**	**(2052 cm)**				
0.015	3.024e+16	1.217e+06	1.389e+06	1.044e+05	1.191e+05	9.113e+04	1.040e+05	9.113e+02	1.040e+03
0.03	1.026e+17	2.706e+07	4.579e+07	2.682e+05	4.538e+05	2.341e+05	3.961e+05	2.341e+03	3.961e+03
0.08	6.188e+16	5.791e+07	1.085e+08	9.165e+04	1.717e+05	8.001e+04	1.499e+05	8.001e+02	1.499e+03
0.1	1.138e+14	1.359e+05	2.347e+05	2.079e+02	3.591e+02	1.815e+02	3.135e+02	1.815e+00	3.135e+00
0.15	3.488e+16	6.451e+07	1.001e+08	1.062e+05	1.648e+05	9.274e+04	1.439e+05	9.274e+02	1.439e+03
0.2	8.110e+16	2.046e+08	2.859e+08	3.611e+05	5.047e+05	3.153e+05	4.406e+05	3.153e+03	4.406e+03
0.3	6.298e+16	2.461e+08	3.171e+08	4.669e+05	6.016e+05	4.076e+05	5.252e+05	4.076e+03	5.252e+03
0.4	1.692e+17	9.018e+08	1.108e+09	1.757e+06	2.159e+06	1.534e+06	1.885e+06	1.534e+04	1.885e+04
0.5	2.849e+17	1.930e+09	2.298e+09	3.788e+06	4.511e+06	3.307e+06	3.938e+06	3.307e+04	3.938e+04
0.6	2.353e+17	1.939e+09	2.260e+09	3.785e+06	4.411e+06	3.304e+06	3.851e+06	3.304e+04	3.851e+04
0.8	4.715e+17	5.286e+09	5.977e+09	1.005e+07	1.137e+07	8.777e+06	9.925e+06	8.777e+04	9.925e+04
1.0	1.950e+17	2.774e+09	3.082e+09	5.114e+06	5.681e+06	4.464e+06	4.960e+06	4.464e+04	4.960e+04
1.5	1.610e+17	3.522e+09	3.809e+09	5.925e+06	6.408e+06	5.173e+06	5.595e+06	5.173e+04	5.595e+04
2.0	1.376e+17	4.078e+09	4.339e+09	6.306e+06	6.710e+06	5.505e+06	5.858e+06	5.505e+04	5.858e+04
3.0	1.244e+16	5.634e+08	5.897e+08	7.643e+05	8.000e+05	6.673e+05	6.984e+05	6.673e+03	6.984e+03
4.0	3.148e+15	1.921e+08	1.992e+08	2.376e+05	2.464e+05	2.075e+05	2.151e+05	2.075e+03	2.151e+03
Totals	**2.044e+18**	**2.179e+10**	**2.452e+10**	**3.913e+07**	**4.431e+07**	**3.416e+07**	**3.868e+07**	**3.416e+05**	**3.868e+05**
	Sensitivity	**Variable**	**Y Dose Point 1**	**(2 of 5)**	**(2802 cm)**				
0.015	3.024e+16	1.219e+06	1.392e+06	1.046e+05	1.194e+05	9.131e+04	1.042e+05	9.131e+02	1.042e+03
0.03	1.026e+17	2.743e+07	4.660e+07	2.719e+05	4.619e+05	2.373e+05	4.032e+05	2.373e+03	4.032e+03
0.08	6.188e+16	5.883e+07	1.104e+08	9.310e+04	1.747e+05	8.127e+04	1.525e+05	8.127e+02	1.525e+03
0.1	1.138e+14	1.380e+05	2.388e+05	2.112e+02	3.653e+02	1.844e+02	3.189e+02	1.844e+00	3.189e+00
0.15	3.488e+16	6.555e+07	1.018e+08	1.079e+05	1.676e+05	9.423e+04	1.464e+05	9.423e+02	1.464e+03
0.2	8.110e+16	2.079e+08	2.907e+08	3.670e+05	5.131e+05	3.204e+05	4.479e+05	3.204e+03	4.479e+03
0.3	6.298e+16	2.501e+08	3.224e+08	4.745e+05	6.116e+05	4.142e+05	5.339e+05	4.142e+03	5.339e+03

0.4	1.692e+17	9.166e+08	1.126e+09	1.786e+06	2.195e+06	1.559e+06	1.916e+06	1.559e+04	1.916e+04
0.5	2.849e+17	1.962e+09	2.336e+09	3.850e+06	4.585e+06	3.361e+06	4.003e+06	3.361e+04	4.003e+04
0.6	2.353e+17	1.971e+09	2.297e+09	3.847e+06	4.484e+06	3.359e+06	3.914e+06	3.359e+04	3.914e+04
0.8	4.715e+17	5.373e+09	6.076e+09	1.022e+07	1.156e+07	8.922e+06	1.009e+07	8.922e+04	1.009e+05
1.0	1.950e+17	2.820e+09	3.133e+09	5.198e+06	5.775e+06	4.538e+06	5.042e+06	4.538e+04	5.042e+04
1.5	1.610e+17	3.580e+09	3.872e+09	6.023e+06	6.515e+06	5.258e+06	5.687e+06	5.258e+04	5.687e+04
2.0	1.376e+17	4.145e+09	4.411e+09	6.410e+06	6.821e+06	5.596e+06	5.955e+06	5.596e+04	5.955e+04
3.0	1.244e+16	5.727e+08	5.994e+08	7.770e+05	8.133e+05	6.783e+05	7.100e+05	6.783e+03	7.100e+03
4.0	3.148e+15	1.953e+08	2.025e+08	2.416e+05	2.505e+05	2.109e+05	2.187e+05	2.109e+03	2.187e+03
Totals	**2.044e+18**	**2.215e+10**	**2.493e+10**	**3.977e+07**	**4.504e+07**	**3.472e+07**	**3.932e+07**	**3.472e+05**	**3.932e+05**
	Sensitivity	**Variable**	**Y Dose Point 1**	**(3 of 5)**	**(3552 cm)**				
0.015	3.024e+16	1.217e+06	1.389e+06	1.044e+05	1.191e+05	9.113e+04	1.040e+05	9.113e+02	1.040e+03
0.03	1.026e+17	2.706e+07	4.579e+07	2.682e+05	4.538e+05	2.342e+05	3.962e+05	2.342e+03	3.962e+03
0.08	6.188e+16	5.792e+07	1.085e+08	9.165e+04	1.717e+05	8.001e+04	1.499e+05	8.001e+02	1.499e+03
0.1	1.138e+14	1.359e+05	2.347e+05	2.079e+02	3.591e+02	1.815e+02	3.135e+02	1.815e+00	3.135e+00
0.15	3.488e+16	6.451e+07	1.001e+08	1.062e+05	1.648e+05	9.275e+04	1.439e+05	9.275e+02	1.439e+03
0.2	8.110e+16	2.046e+08	2.859e+08	3.612e+05	5.047e+05	3.153e+05	4.406e+05	3.153e+03	4.406e+03
0.3	6.298e+16	2.462e+08	3.172e+08	4.669e+05	6.016e+05	4.076e+05	5.252e+05	4.076e+03	5.252e+03
0.4	1.692e+17	9.019e+08	1.108e+09	1.757e+06	2.159e+06	1.534e+06	1.885e+06	1.534e+04	1.885e+04
0.5	2.849e+17	1.930e+09	2.298e+09	3.788e+06	4.511e+06	3.307e+06	3.938e+06	3.307e+04	3.938e+04
0.6	2.353e+17	1.939e+09	2.260e+09	3.785e+06	4.411e+06	3.304e+06	3.851e+06	3.304e+04	3.851e+04
0.8	4.715e+17	5.286e+09	5.977e+09	1.005e+07	1.137e+07	8.778e+06	9.925e+06	8.778e+04	9.925e+04
1.0	1.950e+17	2.774e+09	3.082e+09	5.114e+06	5.682e+06	4.464e+06	4.960e+06	4.464e+04	4.960e+04
1.5	1.610e+17	3.522e+09	3.809e+09	5.926e+06	6.409e+06	5.173e+06	5.595e+06	5.173e+04	5.595e+04
2.0	1.376e+17	4.078e+09	4.339e+09	6.306e+06	6.710e+06	5.505e+06	5.858e+06	5.505e+04	5.858e+04
3.0	1.244e+16	5.634e+08	5.897e+08	7.644e+05	8.001e+05	6.673e+05	6.985e+05	6.673e+03	6.985e+03
4.0	3.148e+15	1.921e+08	1.992e+08	2.377e+05	2.464e+05	2.075e+05	2.151e+05	2.075e+03	2.151e+03
Totals	**2.044e+18**	**2.179e+10**	**2.452e+10**	**3.913e+07**	**4.431e+07**	**3.416e+07**	**3.869e+07**	**3.416e+05**	**3.869e+05**
	Sensitivity	**Variable**	**Y Dose Point 1**	**(4 of 5)**	**(4302 cm)**				
0.015	3.024e+16	1.205e+06	1.373e+06	1.034e+05	1.178e+05	9.024e+04	1.028e+05	9.024e+02	1.028e+03
0.03	1.026e+17	2.574e+07	4.301e+07	2.551e+05	4.263e+05	2.227e+05	3.721e+05	2.227e+03	3.721e+03
0.08	6.188e+16	5.477e+07	1.024e+08	8.667e+04	1.620e+05	7.567e+04	1.414e+05	7.567e+02	1.414e+03
0.1	1.138e+14	1.285e+05	2.217e+05	1.965e+02	3.392e+02	1.716e+02	2.961e+02	1.716e+00	2.961e+00
0.15	3.488e+16	6.098e+07	9.452e+07	1.004e+05	1.557e+05	8.766e+04	1.359e+05	8.766e+02	1.359e+03
0.2	8.110e+16	1.934e+08	2.702e+08	3.413e+05	4.769e+05	2.979e+05	4.163e+05	2.979e+03	4.163e+03
0.3	6.298e+16	2.326e+08	2.997e+08	4.411e+05	5.685e+05	3.851e+05	4.963e+05	3.851e+03	4.963e+03
0.4	1.692e+17	8.520e+08	1.047e+09	1.660e+06	2.040e+06	1.449e+06	1.781e+06	1.449e+04	1.781e+04
0.5	2.849e+17	1.823e+09	2.172e+09	3.579e+06	4.263e+06	3.124e+06	3.721e+06	3.124e+04	3.721e+04
0.6	2.353e+17	1.832e+09	2.135e+09	3.575e+06	4.168e+06	3.121e+06	3.638e+06	3.121e+04	3.638e+04
0.8	4.715e+17	4.993e+09	5.648e+09	9.497e+06	1.074e+07	8.291e+06	9.378e+06	8.291e+04	9.378e+04
1.0	1.950e+17	2.620e+09	2.912e+09	4.830e+06	5.368e+06	4.217e+06	4.686e+06	4.217e+04	4.686e+04
1.5	1.610e+17	3.327e+09	3.599e+09	5.597e+06	6.055e+06	4.886e+06	5.286e+06	4.886e+04	5.286e+04
2.0	1.376e+17	3.852e+09	4.100e+09	5.957e+06	6.340e+06	5.200e+06	5.535e+06	5.200e+04	5.535e+04
3.0	1.244e+16	5.322e+08	5.572e+08	7.220e+05	7.559e+05	6.303e+05	6.599e+05	6.303e+03	6.599e+03
4.0	3.148e+15	1.815e+08	1.882e+08	2.245e+05	2.328e+05	1.960e+05	2.033e+05	1.960e+03	2.033e+03
Totals	**2.044e+18**	**2.058e+10**	**2.317e+10**	**3.697e+07**	**4.187e+07**	**3.227e+07**	**3.655e+07**	**3.227e+05**	**3.655e+05**
	Sensitivity	**Variable**	**Y Dose Point 1**	**(5 of 5)**	**(5052 cm)**				
0.015	3.024e+16	1.133e+06	1.284e+06	9.721e+04	1.101e+05	8.486e+04	9.612e+04	8.486e+02	9.612e+02
0.03	1.026e+17	2.238e+07	3.685e+07	2.218e+05	3.653e+05	1.937e+05	3.189e+05	1.937e+03	3.189e+03
0.08	6.188e+16	4.750e+07	9.004e+07	7.517e+04	1.425e+05	6.562e+04	1.244e+05	6.562e+02	1.244e+03

0.1	1.138e+14	1.114e+05	1.949e+05	1.705e+02	2.982e+02	1.488e+02	2.604e+02	1.488e+00	2.604e+00
0.15	3.488e+16	5.291e+07	8.294e+07	8.712e+04	1.366e+05	7.606e+04	1.192e+05	7.606e+02	1.192e+03
0.2	8.110e+16	1.678e+08	2.368e+08	2.962e+05	4.179e+05	2.586e+05	3.648e+05	2.586e+03	3.648e+03
0.3	6.298e+16	2.020e+08	2.622e+08	3.831e+05	4.974e+05	3.344e+05	4.342e+05	3.344e+03	4.342e+03
0.4	1.692e+17	7.402e+08	9.155e+08	1.442e+06	1.784e+06	1.259e+06	1.557e+06	1.259e+04	1.557e+04
0.5	2.849e+17	1.585e+09	1.898e+09	3.110e+06	3.725e+06	2.715e+06	3.252e+06	2.715e+04	3.252e+04
0.6	2.353e+17	1.592e+09	1.866e+09	3.108e+06	3.642e+06	2.714e+06	3.179e+06	2.714e+04	3.179e+04
0.8	4.715e+17	4.343e+09	4.933e+09	8.261e+06	9.383e+06	7.212e+06	8.191e+06	7.212e+04	8.191e+04
1.0	1.950e+17	2.280e+09	2.543e+09	4.203e+06	4.688e+06	3.669e+06	4.093e+06	3.669e+04	4.093e+04
1.5	1.610e+17	2.897e+09	3.143e+09	4.874e+06	5.288e+06	4.255e+06	4.616e+06	4.255e+04	4.616e+04
2.0	1.376e+17	3.356e+09	3.580e+09	5.190e+06	5.536e+06	4.531e+06	4.833e+06	4.531e+04	4.833e+04
3.0	1.244e+16	4.640e+08	4.866e+08	6.295e+05	6.602e+05	5.496e+05	5.763e+05	5.496e+03	5.763e+03
4.0	3.148e+15	1.583e+08	1.644e+08	1.958e+05	2.034e+05	1.709e+05	1.775e+05	1.709e+03	1.775e+03
Totals	**2.044e+18**	**1.791e+10**	**2.024e+10**	**3.217e+07**	**3.658e+07**	**2.809e+07**	**3.193e+07**	**2.809e+05**	**3.193e+05**

Sensitivity Analysis Summary - Y Dose Point 1										
Dose Point #	**Sensitivity**	**Sensitivity Dimension**	**Fluence Rate MeV/cm²/sec No Buildup**	**Fluence Rate MeV/cm²/sec With Buildup**	**Exposure Rate mR/hr No Buildup**	**Exposure Rate mR/hr With Buildup**	**Absorbed Dose Rate mrad/hr No Buildup**	**Absorbed Dose Rate mrad/hr With Buildup**	**Absorbed Dose Rate mGy/hr No Buildup**	**Absorbed Dose Rate mGy/hr With Buildup**
1	(1 of 5)	(2052 cm)	2.179e+10	2.452e+10	3.913e+07	4.431e+07	3.416e+07	3.868e+07	3.416e+05	3.868e+05
1	(2 of 5)	(2802 cm)	2.215e+10	2.493e+10	3.977e+07	4.504e+07	3.472e+07	3.932e+07	3.472e+05	3.932e+05
1	(3 of 5)	(3552 cm)	2.179e+10	2.452e+10	3.913e+07	4.431e+07	3.416e+07	3.869e+07	3.416e+05	3.869e+05
1	(4 of 5)	(4302 cm)	2.058e+10	2.317e+10	3.697e+07	4.187e+07	3.227e+07	3.655e+07	3.227e+05	3.655e+05
1	(5 of 5)	(5052 cm)	1.791e+10	2.024e+10	3.217e+07	3.658e+07	2.809e+07	3.193e+07	2.809e+05	3.193e+05


Cont. Rad Response
Dose Point 1 - (2287,2802.6,0) cm
1.0e+11
1.0e+10
photons/cm²/sec
0
1000
2000
3000
4000
5000
6000
Y Dose Point 1 (cm)
Without Buildup
With Buildup

ENERCON Excellence—Every project. Every day.	CALCULATION COVER SHEET	CALC. NO. STPNOC013-CALC-006
		REV. 3
		PAGE NO. 1 of 50

Title:	Dose Rate Evaluation of Reactor Vessel Water Levels during Refueling for EAL Thresholds	**Client:** STP
		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:

Revision 1 added reference for reactor vessel head thickness, and updated calculations with new value (7.19 in). Removed any detector specific calculations so results can be applied to any detector at these locations. Made several editorial changes.

Revision 2 changed the Safety-Related designation to Non-Safety Related. Minor grammatical edits.

Revision 3 added a decay time to the source term, modeled the upper internals for the cases with reactor vessel head attached, made minor changes to the model geometry, and minor editorial changes.

Revision Impact on Results:

For Revision 1, the dose rates for the cases with reactor vessel head attached are higher due to the reduction in head thickness.

For Revision 2, there was no impact on results.

For Revision 3, the dose rates are lower due to the decay time of the source term and modeling of the upper internals.

Study Calculation ☐	**Final Calculation** ☒
Safety-Related ☐	**Non-Safety Related** ☒

(Print Name and Sign)

Originator:	**Jay Bhatt / Zachary Rose**	**Date:**	6/29/2015
Reviewer:	**Adam Wright**	**Date:**	6/29/2015
Approver:	JMaisler Digitally signed by Jay Maisler DN: cn=Jay Maisler, o=New Plant Services, ou=Manager of Projects, email=jmaisler@enercon.com, c=US Date: 2015.06.29 17:45:01 -04'00'	**Date:**	6/29/2015

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
0	02/07/2014	Initial Issue
1	03/21/2014	Updated containment dimensions including reactor vessel head thickness. Added more detail to calculations section. Made editorial changes.
2	2/17/2015	Change Safety-Related designation.
3	6/29/2015	Added cool time and modeled upper internals. Updated design inputs and geometry.

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION
1-43	3		

APPENDIX REVISION STATUS

APPENDIX NO.	PAGE NO.	REVISION NO.	APPENDIX NO.	PAGE NO.	REVISION NO.
A	44	3			
B	45-50	3			

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List of Tables

1. Purpose and Scope

The purpose of this calculation is to evaluate dose rates as a function of water height in the reactor vessel during refueling operations in order to set Emergency Action Level (EAL) thresholds for core uncovery. The dose rates are calculated at the locations of the containment monitors RE-8055 and RE-8099 so that dose rate measurements by these devices can be used to estimate water level in the core, upon failure of other water level detection systems. This evaluation will calculate the dose rate at full core uncovery, as well as maximum water levels with a detectable dose rate response. Since the scope of this calculation concerns uncovering the reactor core, the effects of future fuel element storage in the nearby Fuel Storage Pit are not analyzed, as its effects are negligible in comparison. The containment building, components within the building, and the reactor vessel and contents are modeled simplistically because only order of magnitude results are needed. As such, the dose rate results should be considered as reasonably representative of the magnitude of the actual dose rate only.

2. Summary of Results and Conclusion

The dose rate results for the configuration without the reactor vessel head and with the reactor vessel head are provided in Section 7.8.1 and Section 7.8.2, respectively. The maximum dose rates with the core uncovered (i.e. water at the top of the active length) are shown in the table below.

Model Description	Dose Rate (mrem/h)
Head On (Vessel Internals Modeled)	1.51E-04
Head On (Vessel Internals not Modeled)	3.16E+03
Head Off	4.75E+06
Head Off (Appendix B)	3.00E+05

Detailed results of the dose rate as a function of water height are provided in Figure 7-13 with the head removed and Figure 7-14 with the head attached.

Appendix B contains the results of a sensitivity case that is run to evaluate additional geometric changes and the effect of including the Upper Fuel Hardware region for the No-Head- 0, 2 and 4 ft. above top of active fuel water level cases.

3. References

1. "Standard Composition Library," ORNL/NUREG/CSD-2/V1/R6, Volume 3, Section M8, March 2000.
2. Calculation NC-6510, Rev. 0, "Core Radionuclide Inventory for Chapter 15 Accident Analysis."
3. RSICC Code Package CCC-785, "A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design," July 2011.
4. "ORIGEN-S: Depletion Module to Calculate Neutron Activation, Actinide Transmutation, Fission Product Generation, and Radiation Source Terms", I. C. Gauld, January 2011.
5. Drawing 3C01-1-C-1509, Rev. 16, Concrete Reactor Containment Building Internal – Plan @ El 19'-0" Unit No. 1
6. Drawing 3C01-2-C-1521, Rev. 12, Concrete Reactor Containment Building Internal – Plan el 68'-0" Unit No. 2
7. Drawing 6C-18-N-5006, Rev. 9. "General Arrangement Reactor Containment Building Plan at El. 68' 0" Area G."
8. Drawing 6C-18-9-N-5007, Rev. 6. "General Arrangement Reactor Containment Building Section A-A Area G."
9. Drawing 6C-18-9-N-5008, Rev. 8. "General Arrangement Reactor Containment Building Section B-B Area G."
10. RSICC Code Package CCC-810, "MCNP6.1/MCNP5/MCNPX Monte Carlo N–Particle Transport Code System Including MCNP6.1, MCNP5-1.60, MCNPX-2.7.0 and Data Libraries," October 2008.
11. ANSI/ANS 6.1.1-1977, Neutron and Gamma Flux-To-Dose Conversion Factors.
12. Drawing 3C01-1-C-1522, Rev. 17, Concrete Reactor Containment Building Internal Plan @ El. 18'-0" Unit No. 1
13. Drawing L5-01EM101, Rev. 1. "Closure Head General Assembly."
14. Drawing 1142E24. "Model 4XLR Reactor 173 in. I.D. Vessel."
15. Drawing 2C26-9-S-1004, Rev. 4. "Steel Reactor Containment Building Cylindrical Shell Liner Sects. And Dets. Unit No. 1 & 2."
16. Drawing 1211E6. "4 Loop Rapid XL Reactor General Assembly."
17. Letter NOCXX15028833, STP Reactor Vessel Internals Shielding Model Input
18. STP Technical Requirements Manual Section 3.9.3, Amendment 76.
19. Drawing 5C159Z00222, Rev. 7. "Instrument Piping Reactor Containment Building Plan at El. 68'-0"."
20. Drawing 5C159Z00224, Rev. 9. "Instrument Piping Reactor Containment Building Plan at El. 68'-0"."

4. Assumptions

The following assumptions are used in the core uncovery dose rate calculation:

1. The core is homogenized based on the typical Westinghouse 17x17 fuel assembly dimensions, taking into account the fuel rods and space between. Any small variations in fuel parameters will have a negligible effect on containment dose rates. The cladding is modeled as Zircaloy 4 in lieu of ZIRLO; this is standard practice in nuclear fuel shielding evaluations.
2. Any non-fuel hardware is ignored in the active fuel region, since the primary self-shielding occurs in the fuel itself, and there may be some unknown streaming effects through the non-fuel hardware. This homogenization takes into account the presence of water calculating the isotopic weight fraction and homogenized density. For the cases with the reactor vessel head attached, the regions between the head and the active fuel region are homogenized based on the actual mass of the of the upper internals and the modeled water level. Homogenization of source regions and shields is a standard practice in nuclear fuel shielding evaluations.
3. The compositions of the containment structure and components are based on the values in the SCALE standard composition library [1]. This is standard practice in nuclear fuel shielding evaluations.
4. The containment outer concrete thickness is modeled as 3 feet thick. Because the backscattering off the containment walls is due to the steel liner, this dimension has a negligible impact on dose rates near the reactor vessel.
5. Irradiated fuel cannot be moved prior to 100 hours of reactor subcriticality per LCO 3.9.3 [18]. This calculation assumes a decay time of 50 hours to allow EAL thresholds to be determined for reactor vessel conditions that exist prior to the commencement of fuel movement. Due to the order of magnitude of the dose rates involved, the fuel gamma source term will predominate and the N-gamma and hardware activation can be neglected.
6. The elevation of detectors is modeled at the Unit 2 elevation. The difference between the Unit 1 and Unit 2 detector elevations is negligible based on distance from the reactor vessel.
7. Particles entering the steam generators are assumed to remain in the steam generator; therefore, a thickness of 20 cm is used to model the steam generator outer shell.
8. The hardware in the upper internals region between the active fuel region and reactor vessel head is assumed to be stainless steel type 304. While the actual composition of the hardware may vary slightly, small variations in the material will have a negligible effect on the dose rate response at the detectors.

5. Design Inputs

5.1 Fuel Assembly Parameters

The following fuel assembly parameters are used in the core homogenization in the MCNP model. They are based on typical fuel assembly values for Westinghouse 17x17 fuel.

Table 5-1 Design Input Fuel Assembly Parameters for Westinghouse Fuel

Parameters	Value	Unit	Reference
Fuel Type	Westinghouse 17x17		Assumption 1
# Fuel Rods per Assy	264		Assumption 1
Assembly Array	17x17		Assumption 1
Assembly Width	8.404	[in]	Assumption 1
Enrichment	4	wt %	Assumption 1
Density (% of theoretical)	0.95		Assumption 1
Fuel Pellet OD	0.3225	[in]	Assumption 1
Fuel Rod Pitch	0.496	[in]	Assumption 1
Fuel Rod OD	0.374	[in]	Assumption 1
Clad Thickness	0.0225	[in]	Assumption 1
Guide Tube OD	0.482	[in]	Assumption 1
Guide Tube Thickness	0.020	[in]	Assumption 1
# Guide Tubes	24		Assumption 1
Instrument Tube OD	0.482	[in]	Assumption 1
Instrument Tube Thickness	0.020	[in]	Assumption 1
# Instrument Tubes	1		Assumption 1
Active Length	14	[ft]	[17]

5.2 Containment Dimensions

The following dimensions are based on drawings of the STP containment building and equipment. Some parameters are estimated using scaling when the drawings do not detail the exact dimension. These estimations are only applied to dimensions that have a negligible effect on the core uncovery dose rate analysis.

Note: All elevations are relative to the bottom of the active fuel elevation of 12'-1" (368.3 cm) [9]. Dimensions with an asterisk indicate a change was evaluated in Appendix B.

Table 5-2 Design Input Containment Dimensions

Dimension:	ft.	in	cm	reference
Reactor Pressure Vessel (RPV)				
Elevation at Head Level Platform	26	5 ½	806.45	[8]
Elevation at Full Water Level in Refueling Cavity	54	5	1658.62	[8]
Closure Head Thickness	0	7.19	18.26	[17]
RPV Inner Diameter at Shell	14	5	439.42	[14]
RPV Thickness	0	10	25.40	[14] Scaled
RPV Outer Diameter			490.22	Calculated
Thickness of Concrete around RPV*	16	4	497.84	[9] Scaled
Steam Generator (SG)				
Lower Modeled Elevation	26	3	800.10	[9]
Upper Modeled Elevation	93	8 7/8	2857.18	[9]
Overall Modeled Height			2057.08	Calculated
SG Outer Diameter	16	5	500.38	[7] Scaled
SG Distance from RPV (x-plane)	12	5 1/8	378.78	[7]
SG Distance from RPV (y-plane)	30	9 1/8	937.58	[7]
SG Thickness	0	7 7/8	20.00	Assumption 7
Active Fuel				
Elevation at Top of Active Fuel (Bot. of Fuel + Fuel Height)	14	0	426.72	Calculated
Diameter of Active Fuel	12	8 1/2	387.35	[16]
Concrete Wall				
Lower Elevation	26	5 1/2	806.45	[9]
Upper Elevation	70	11	2161.54	[9]
Overall Height			1355.09	Calculated
Thickness	3	6	106.68	[7] Scaled
Inner Width*	35	6	1082.04	[7] Scaled

Dimension:	ft.	in	cm	reference
Inner Length*	100*	0	3048.00	[7] Scaled
Primary Loop Void Lower Elevation	17	3	525.78	[8]
Primary Loop Void Net Height	7	0	213.36	[8] Scaled
Primary Loop Void Upper Elevation (Lower El. + Height)			739.14	Calculated
Primary Loop Void Width	4	6	137.16	[8] Scaled
Thickness of Concrete Between Void and Lower Elevation (Lower El. – Void Upper El.)			67.31	Calculated
Thickness between RPV and Void	1	0	30.48	[8] Scaled
Inner Diameter of Void from RPV (RPV Outer Dia. + Thickness Between Void and RPV)			551.18	Calculated
Outer Diameter of Void from RPV (Inner Dia. of Void + Void Width)			825.50	Calculated
Containment				
Upper Modeled Elevation	140	11	4295.14	[8]
Lower Modeled Elevation	55	11	1704.34	[8]
Net Height			2590.80	Calculated
Inner Diameter	149	11 1/4	4570.10	[15]
Liner Thickness	0	3/8	0.95	[15]
Dome Inner Radius	74	11 5/8	2285.05	[15]
Concrete Thickness	3	0	91.44	Assumption 4
Detector Locations				
Detector RE-8055 Elevation	60	11	1856.74	[19]
Detector RE-8055 Distance from RPV (x-plane) *	36	6	1112.52	[7] and [19] Scaled
Detector RE-8055 Distance from RPV (y-plane) *	17	9	541.01	[7] and [19] Scaled

* Modeled at 100 ft to ensure no overlap with the modeled detectors.

Dimension:	ft.	in	cm	reference
Detector RE-8099 Elevation	60	11	1856.74	[20]
Detector RE-8099 Distance from RPV (x-plane) *	48	0	1463.04	[7] and [20] Scaled
Detector RE-8099 Distance from RPV (y-plane) *	17	9	541.01	[7] and [20] Scaled

5.3 Core Isotopic Inventory

Core isotopic activities are provided in Table 12 of [2]. The isotope specific activities are given in terms of Ci/MWt, which is converted to curies based on the total core thermal power of 4,100 MWt [2]. These calculations are performed in EXCEL spreadsheet *STP.xlsx*. A table of the input values is shown in Table 5-3, below.

Table 5-3 Design Basis Core Shutdown Source Term†

Isotope	Ci/MWt	Ci	Isotope	Ci/MWt	Ci
Kr83m	3.41E+03	1.40E+07	Ru106	1.34E+04	5.49E+07
Kr85m	7.07E+03	2.90E+07	Rh105	3.05E+04	1.25E+08
Kr85	2.93E+02	1.20E+06	Zr95	4.39E+04	1.80E+08
Kr87	1.34E+04	5.49E+07	Zr97	4.39E+04	1.80E+08
Kr88	1.90E+04	7.79E+07	Nb95	4.32E+04	1.77E+08
Kr89	2.32E+04	9.51E+07	La140	4.63E+04	1.90E+08
Xe131m	2.68E+02	1.10E+06	La141	4.62E+04	1.89E+08
Xe133m	1.66E+03	6.81E+06	La142	4.15E+04	1.70E+08
Xe133	5.37E+04	2.20E+08	Pr143	3.90E+04	1.60E+08
Xe135m	1.02E+04	4.18E+07	Nd147	1.73E+04	7.09E+07
Xe135	1.34E+04	5.49E+07	Am241	2.75E+00	1.13E+04
Xe137	4.63E+04	1.90E+08	Cm242	1.05E+03	4.31E+06
Xe138	4.39E+04	1.80E+08	Cm244	6.17E+01	2.53E+05
I131	2.59E+04	1.06E+08	Ce141	4.39E+04	1.80E+08
I132	3.71E+04	1.52E+08	Ce143	4.15E+04	1.70E+08
I133	5.37E+04	2.20E+08	Ce144	3.41E+04	1.40E+08
I134	5.85E+04	2.40E+08	Np239	5.12E+05	2.10E+09
I135	4.88E+04	2.00E+08	Pu238	8.71E+01	3.57E+05
Sb127	3.05E+03	1.25E+07	Pu239	1.96E+01	8.04E+04
Sb129	8.29E+03	3.40E+07	Pu240	2.48E+01	1.02E+05
Te127m	4.32E+02	1.77E+06	Pu241	4.17E+03	1.71E+07
Te127	3.05E+03	1.25E+07	Rb86	9.92E+01	4.07E+05
Te129m	1.22E+03	5.00E+06	Cs134	5.37E+03	2.20E+07
Te129	8.05E+03	3.30E+07	Cs136	1.54E+03	6.31E+06
Te131m	3.66E+03	1.50E+07	Cs137	3.17E+03	1.30E+07
Te132	3.82E+04	1.57E+08	Y90	3.56E+03	1.46E+07
Ba137m	2.93E+03	1.20E+07	Y91	3.41E+04	1.40E+08
Ba139	4.98E+04	2.04E+08	Y92	3.41E+04	1.40E+08
Ba140	4.63E+04	1.90E+08	Y93	3.90E+04	1.60E+08
Mo99	4.83E+04	1.98E+08	Sr89	2.68E+04	1.10E+08
Tc99m	4.07E+04	1.67E+08	Sr90	2.37E+03	9.72E+06
Ru103	3.90E+04	1.60E+08	Sr91	3.17E+04	1.30E+08
Ru105	2.68E+04	1.10E+08	Sr92	3.41E+04	1.40E+08

† Ci = Ci/MWt × 4,100 MWt

5.4 Material Compositions

The following compositions used in the MCNP model are taken from the SCALE standard composition library [1] and are shown in Table 5-4.

Table 5-4 SCALE Standard Compositions used in MCNP Model

Material	Isotope	Weight Fraction	Reference
Zry- 4	Zr	0.9823	[1]
(6.56 g/cm^3)	Sn	0.0145	
	Cr	0.0010	
	Fe	0.0021	
	Hf	0.0001	
UO_2	U-235	0.0353	[1]
(10.412 g/cm^3)‡	U-238	0.8462	
	O	0.1186	
Air	C	0.0001	[1]
(1.21E-03 g/cm^3)	N	0.7651	
	O	0.2348	
Water	H	0.1111	[1]
(0.9982 g/cm^3)	O	0.8889	
SS-304	Fe	0.6838	[1]
(7.94 g/cm^3)	Cr	0.1900	
	Ni	0.0950	
	Mn	0.0200	
	Si	0.0100	
	C	0.0008	
	P	0.0004	
Concrete	O	0.5320	[1]
(2.30 g/cm^3)	Si	0.3370	
	Ca	0.0440	
	Al	0.0340	
	Na	0.0290	
	Fe	0.0140	
	H	0.0100	
Carbon Steel	C	0.0100	[1]
(7.82 g/cm^3)	Fe	0.9900	

‡ Based on 95% of theoretical density, Assumption 1.

5.5 Upper Internals/Upper Fuel Hardware

The following are used in the MCNP model for the Upper Internals/Upper Fuel Hardware Region [17]:

- The Upper Fuel Hardware region elevation range is from 426.72 cm to 465.90 cm.
- The Upper Fuel Hardware region total metal mass is 3860 kg.
- The Upper Internals region elevation range is from 465.90 cm to 806.26 cm.
- The Upper Internals total metal mass is 61870 kg.

6. Methodology

The reactor source terms are computed with ORIGEN-S of the SCALE 6.1 code package [3, 4]. The ORIGEN-S decay sequence is used to bin design input isotope specific activities into energy dependent photon bins. These energy specific photon emission bins are used as input for the energy distribution described by the MCNP source definitions.

The MCNP6 [10] Monte Carlo transport code is used to determine the dose rates.

The detailed engineering drawings are converted into MCNP surface and cell cards in the proper dimensions. The radiation monitors of interest are modeled as point detectors to determine the expected dose rate for those detectors. The dose rates are calculated as a function of water height for two reactor refueling conditions:

1. With Head – the reactor is modeled with a 7.19 inch carbon steel plate as indicated in Table 5-2, which is additional attenuation between source and detector. The mass of the Upper Internals and Upper Fuel Hardware including top nozzle, core support plate, and upper guide structure are homogenized into discrete regions between the active fuel region and the vessel head.

2. Without head – the reactor is modeled with nothing between the active fuel zone and containment. Note that Appendix B includes the Upper Fuel Hardware region above the active fuel zone.

Variance reduction is accomplished with a geometric importance map that is imposed on the homogenized core. In addition, cell based weight windows are utilized to improve the variance reduction of the simple geometric scheme. A superimposed weight window mesh is utilized where necessary to improve variance. The weight windows are iteratively generated using the MCNP weight windows generator card. All final dose rates presented in this calculation include weight windows variance reduction.

7. Calculations

7.1 Source Terms

In order to convert the isotope specific activity into an energy spectrum, ORIGEN-S of the SCALE6.1 code package is used to initiate a decay and bin into 19 photon energy groups. The energy groups along with their associated activities are used in the MCNP source definition to model the anticipated radiation emission following shutdown.

The ORIGEN-S input deck, *STPEALa.inp*, is provided below in Figure 7-1. This input has a simple decay case where the inputted isotopic composition in curies is decayed. The isotope is specified in the 73$$ card using the special identifier described in Section F7.6.2 of the ORIGEN-S manual, and the activity in curies is specified in the 74** card. The time steps for the decay are given on the 60** card in hours. Although multiple time steps are calculated, the source term with 50 hours decay time is used in this calculation to model the core shortly after shutdown. The output of the decay is given in terms of photons/s/Energy-Group, which is automatically normalized in the MCNP input.

Figure 7-1 ORIGEN-S Input Deck for MCNP Source Term Calculation

```
=origens                                                         ←Call Origen-S Sequence
0$$ a11 71 e t                                                   ←Logical Unit Assignments
                                                                        -Binary Photon Library (71)
PWR Source Term STP ELA Analysis                                 ←Case Title
3$$  21 1 1 a4 27 a16 4 a33 19 e t                               ←Library Integer Constants
                                                                        -Units 83** Card Ci (4)
                                                                        -Gamma Energy Groups (19)

35$$ 0 t                                                         ←Not Used
54$$ a8 0 a11 2 e                                                ←Special Calculation Options
                                                                        -Cutoff Value (Default)
                                                                        -(α,n) Composition Dependent

56$$ 0 6 a6 1 a10 0 a13 66 3 3 0 2 0 e                           ←Subcase Control Constants
                                                                        -Decay Only Subcase (0)
                                                                        -Number of Time Intervals (6)
                                                                        -Number of Nuclides (66)
                                                                        -Unit of Time in Hours (3)
57** 0  a3 1-16 e                                                ←Not Used
95$$ 0 t                                                         ←Not Used
STPEAL                                                           ←Subcase Title
Ci Source Terms                                                  ←Subcase Basis
60** 0 24 40 50 60 70                                            ←Time (hours)
61** 5r1-8 1+6 1+4                                               ←Cutoff Values
65$$                                                             ←Decay Period Print Triggers
'GRAM-ATOMS   GRAMS     CURIES      WATTS-ALL  WATTS-GAMMA
     3Z     0  1  0     1 0 0       1 0 0         3Z         6Z
     3Z     1  1  1     1 0 1       1 1 1         3Z         6Z
     3Z     1  1  1     1 1 1       1 1 1         3Z         6Z
81$$ 2 0 26 1 e                                                  ←Gamma Source Constants
82$$ f2                                                          ←Produces Gamma Source Spectrum
83** 1.10E+07 1.00E+07 8.00E+06 6.50E+06 5.00E+06 4.00E+06 3.00E+06 ←Gamma Energy Groups
     2.50E+06 2.00E+06 1.66E+06 1.33E+06 1.00E+06 8.00E+05 6.00E+05
     4.00E+05 3.00E+05 2.00E+05 1.00E+05 5.00E+04 1.00E+04 e
84** 2.00E+07 6.43E+06 3.00E+06 1.85E+06 1.40E+06 9.00E+05 4.00E+05 ←Neutron Energy Groups
     1.00E+05 1.70E+04 3.00E+03 5.50E+02 1.00E+02 3.00E+01 1.00E+01    (Not Used)
     3.05E+00 1.77E+00 1.30E+00 1.13E+00 1.00E+00 8.00E-01 4.00E-01
     3.25E-01 2.25E-01 1.00E-01 5.00E-02 3.00E-02 1.00E-02 1.00E-05 e
73$$ 360831 360851 360850 360870 360880 360890 541311 541331 ←Nuclide Identifiers
    541330 541351 541350 541370 541380 531310 531320 531330
    531340 531350 511270 511290 521271 521270 521291 521290
    521311 521320 561371 561390 561400 420990 430991 441030
    441050 441060 451050 400950 400970 410950 571400 571410
    571420 591430 601470 952410 962420 962440 581410 581430
    581440 932390 942380 942390 942400 942410 370860 551340
    551360 551370 390900 390910 390920 390930 380890 380900
    380910 380920
74** 1.40E+07 2.90E+07 1.20E+06 5.49E+07 7.79E+07 9.51E+07 1.10E+06 ←Nuclide Concentrations (Ci)
    6.81E+06 2.20E+08 4.18E+07 5.49E+07 1.90E+08 1.80E+08 1.06E+08
    1.52E+08 2.20E+08 2.40E+08 2.00E+08 1.25E+07 3.40E+07 1.77E+06
    1.25E+07 5.00E+06 3.30E+07 1.50E+07 1.57E+08 1.20E+07 2.04E+08
    1.90E+08 1.98E+08 1.67E+08 1.60E+08 1.10E+08 5.49E+07 1.25E+08
    1.80E+08 1.80E+08 1.77E+08 1.90E+08 1.89E+08 1.70E+08 1.60E+08
    7.09E+07 1.13E+04 4.31E+06 2.53E+05 1.80E+08 1.70E+08 1.40E+08
    2.10E+09 3.57E+05 8.04E+04 1.02E+05 1.71E+07 4.07E+05 2.20E+07
    6.31E+06 1.30E+07 1.46E+07 1.40E+08 1.40E+08 1.60E+08 1.10E+08
    9.72E+06 1.30E+08 1.40E+08
75$$ 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 ←Library Kind
                                                                        2-Actinide 3-Fission Product
    3 3 3 3 3 3 3 3 3 3 3 2 2 2 3 3 3 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3
    3 3
t
56$$ f0 t
End
```

The results of this calculation are summarized below in Table 7-1. These values will be used in the MCNP input source definition.

Table 7-1 Binned Total Core Source Term

Energy Group	Energy Boundaries (MeV)	Photons/sec
1	0.01-0.05	4.040E+19
2	0.05-0.1	1.306E+19
3	0.1-0.2	3.045E+19
4	0.2-0.3	1.893E+19
5	0.3-0.4	7.253E+18
6	0.4-0.6	1.511E+19
7	0.6-0.8	2.766E+19
8	0.8-1	4.457E+18
9	1-1.33	1.016E+18
10	1.33-1.66	7.389E+18
11	1.66-2	1.391E+17
12	2-2.5	1.558E+17
13	2.5-3	2.286E+17
14	3-4	1.785E+15
15	4-5	3.827E+10
16	5-6.5	1.040E+09
17	6.5-8	1.741E+08
18	8-10	3.695E+07
19	10-11	2.001E+06
totals		1.662E+20

7.2 MCNP Model Core Homogenization

Because the source term is given for the entire core, the self-shielding from the assemblies is an important part of the dose rate response in regions above the core. Particles born in the lower section of the core are very unlikely to penetrate through the core itself, and make it to the radiation monitors. For simplicity, the core is modeled as a 3 dimensional cylinder with a uniformly distributed spatial particle distribution. The calculations for the homogenization are done in the worksheet *Compositions* of the EXCEL workbook *STP.xlsx*. A density and isotopic composition is calculated with the water level above the top of the fuel. A summary of the calculations for the core composition and density is shown below. Note that the EXCEL workbook uses additional significant figures. The inputs are based on the dimensions in Table 5-1 and the compositions in Table 5-4.

$$Rod\ Volume = \pi(Pellet\ Radius)^2 \times Active\ Length = \pi(0.16125\ in)^2(168\ in) = 13.72\ in^3$$

$$Rod\ Mass_{UO_2} = \rho \times V = \left(10.412\frac{g}{cc}\right)(13.72\ in^3)\left(2.54\frac{cm}{in}\right)^3 = 2340.9\ g$$

$$Assembly\ Mass_{UO_2} = Rod\ Mass \times \frac{Number\ of\ Fuel\ Rods}{Assembly} = (2340.9\ g)(264) = 618.0\ kg$$

$$Clad\ Volume = \pi\left(\frac{OD^2}{4} - \frac{ID^2}{4}\right) \times Active\ Length = (3.14)\left[\frac{(0.374\ in)^2}{4} - \frac{(0.329\ in)^2}{4}\right](168\ in)$$
$$= 4.17\ in^3$$

$$Rod\ Mass_{Zry-4} = \rho \times V = \left(6.56\ \frac{g}{cc}\right)(4.17\ in^3)\left(2.54\frac{cm}{in}\right)^3 = 448.3\ g$$

$$Assembly\ Mass_{Zry-4} = Rod\ Mass \times \frac{Number\ of\ Fuel\ Rods}{Assembly} = (448.3\ g)(264) = 118.4\ kg$$

$$Assembly\ \ H_2O\ Volume$$
$$= [(Assembly\ Width)^2 - \pi(Rod\ Radius)^2 \times Number\ of\ Fuel\ Rods]$$
$$\times Active\ Length = [(8.404\ in)^2 - (3.14)(0.187\ in)^2(264)](168\ in) = 6993\ in^3$$

$$Assembly\ Mass_{H_2O} = \rho \times V = \left(0.9982\frac{g}{cc}\right)(6993\ in^3)\left(2.54\frac{cm}{in}\right)^3 = 114.4\ kg$$

$$Assembly\ Volume = Active\ Length\ \times (Assembly\ Width)^2 = (168\ in)(8.404\ in)^2 = 11865.4\ in^3$$

$$Density = \frac{Total\ Mass}{Volume} = \frac{1000g/kg(618.0 + 118.4 + 114.4)\ kg}{11865.4\ in^3\left(2.54\frac{cm}{in}\right)^3} = 4.38\ g/cc$$

The corresponding isotopic composition for the homogenized active fuel region is calculated as seen in the table below.

ZAID Number	Atom	Mass Fraction Active Fuel Region *Homogenized*
92235	U-235	0.0256
92238	U-238	0.6146
8016	O	0.2056
40000	Zr	0.1367
50000	Sn	0.0020
24000	Cr	0.0001
26000	Fe	0.0003
72000	Hf	0.0000
1001	H	0.0149

7.3 MCNP Model Upper Internals Homogenization

For the cases with the RPV head attached, the Upper Internals (UI) and Upper Fuel Hardware (UFH) region are modeled as five discrete cylinders with uniformly distributed homogenized material to account for the different regions between the active fuel height and RPV head. The homogenization accounts for the mass of metal from Section 5.5 (assumed stainless steel type 304 per Assumption 8) and the mass of water (where applicable). The calculations for the homogenization are done in the EXCEL workbook *STP Homogeneous.xlsx*. The mass of stainless steel for the Upper Internals and Upper Fuel Hardware is divided into each of the five cylinders in order to determine the effective density of stainless steel for each of the volumes. If the region above the core is not to be modeled with water, the corresponding cell is defined as stainless steel at its calculated effective density, as seen in the table below.

Cell	Height (cm)	Region	Mass (g)	Volume (cm^3)	Steel Density (g/cm^3)	Homogenized Density (g/cm^3)
45	135.89	UI	26374861	20597608	1.2805	n/a
44	60.96	UI	11831713	9240049	1.2805	2.1177
43	60.96	UI	11831713	9240049	1.2805	2.1177
42	60.96	UI	11831713	9240049	1.2805	2.1177
41	60.96*	UFH	3860000	9240049	0.4177	1.3634

*Note that the height of the UFH given in the design input is 39.18 cm, however the mass of stainless steel is homogenized over a height of 60.96 cm in cell 41 for simplicity in order to model the water level at 2 feet increments above the active fuel height. As the model conserves the mass of the Upper Fuel Hardware region, the shielding properties of the cell are adequate.

If the region is to be modeled with water, the corresponding mass of water for each of the cylindrical volumes is determined by subtracting the volume occupied by the mass of steel. The homogenized density shown in the table above is calculated assuming the remainder of the cell is filled with water. The corresponding isotopic composition for the homogenized cells full of water are calculated for the applicable regions, as seen in the table below.

ZAID Number	Atom	Mass Fraction Steel and Water Homogenized	
		Cell 41/UFH	Cells 42-44/UI
1001	H	7.71E-02	4.39E-02
8016	O	6.17E-01	3.51E-01
26000	Fe	2.09E-01	4.13E-01
24000	Cr	5.82E-02	1.15E-01
28000	Ni	2.91E-02	5.74E-02
25055	Mn	6.13E-03	1.21E-02
14000	Si	3.06E-03	6.05E-03

ZAID Number	Atom	Mass Fraction Steel and Water Homogenized	
		Cell 41/UFH	Cells 42-44/UI
6000	C	2.45E-04	4.84E-04
15031	P	1.38E-04	2.72E-04

In addition, two cases are run with the RPV head attached that neglect this homogenized mass of stainless steel; these cases are run with the water level at 0 feet and 8 feet above the active fuel height. For these cases the materials between the active fuel height and the RPV head are considered to be water or air only.

7.4 MCNP Model Geometry

The following MCNP model geometry is based on the containment dimensions summarized in Table 5-2. The model only focuses on the primary systems and components that provide shielding or reflection from the core to the radiation monitors. These components include the reactor vessel, concrete in reactor pit, containment walls (reflection), and steam generators (reflection). VISED plots of the model geometry are provided in Figure 7-2, Figure 7-3, and Figure 7-4. The MCNP surface cards with the model dimensions (cm) are shown in Figure 7-5, and the cell cards are shown in Figure 7-6 for the cases with no reactor head. A VISED plot of the model with the reactor head is shown in Figure 7-7. The cell cards for the cases with the reactor head are shown in Figure 7-8. Areas that are not of interest are given an importance of zero (white areas) so MCNP will not track particles in locations that will not contribute to the detector response. A summary of surfaces used in constructing this geometry is shown in Table 7-2, including a description of macrobody dimensions.

Figure 7-2 X-Z VISED Plot of Reactor Vessel and Concrete Reactor Pit (No Head)

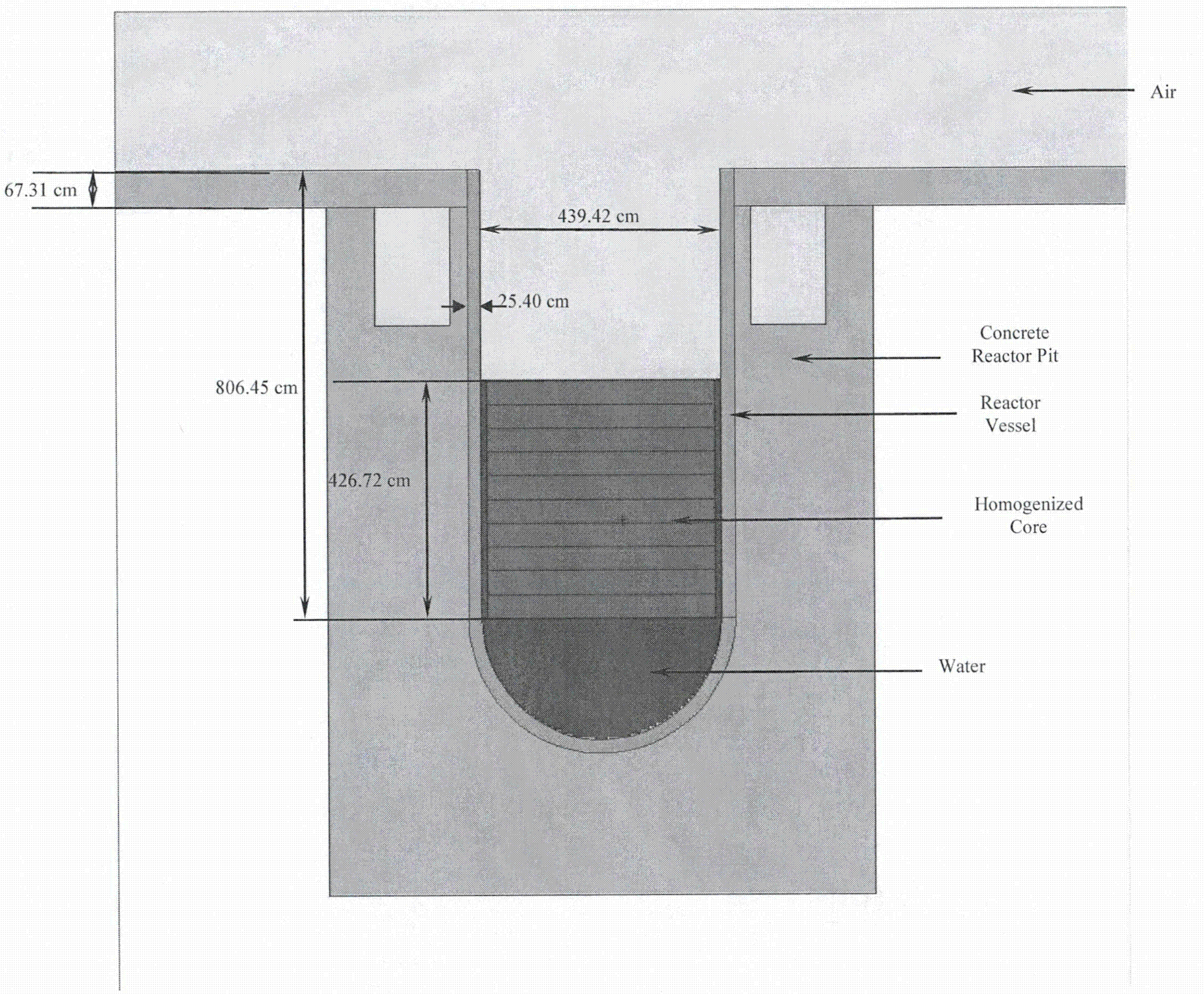
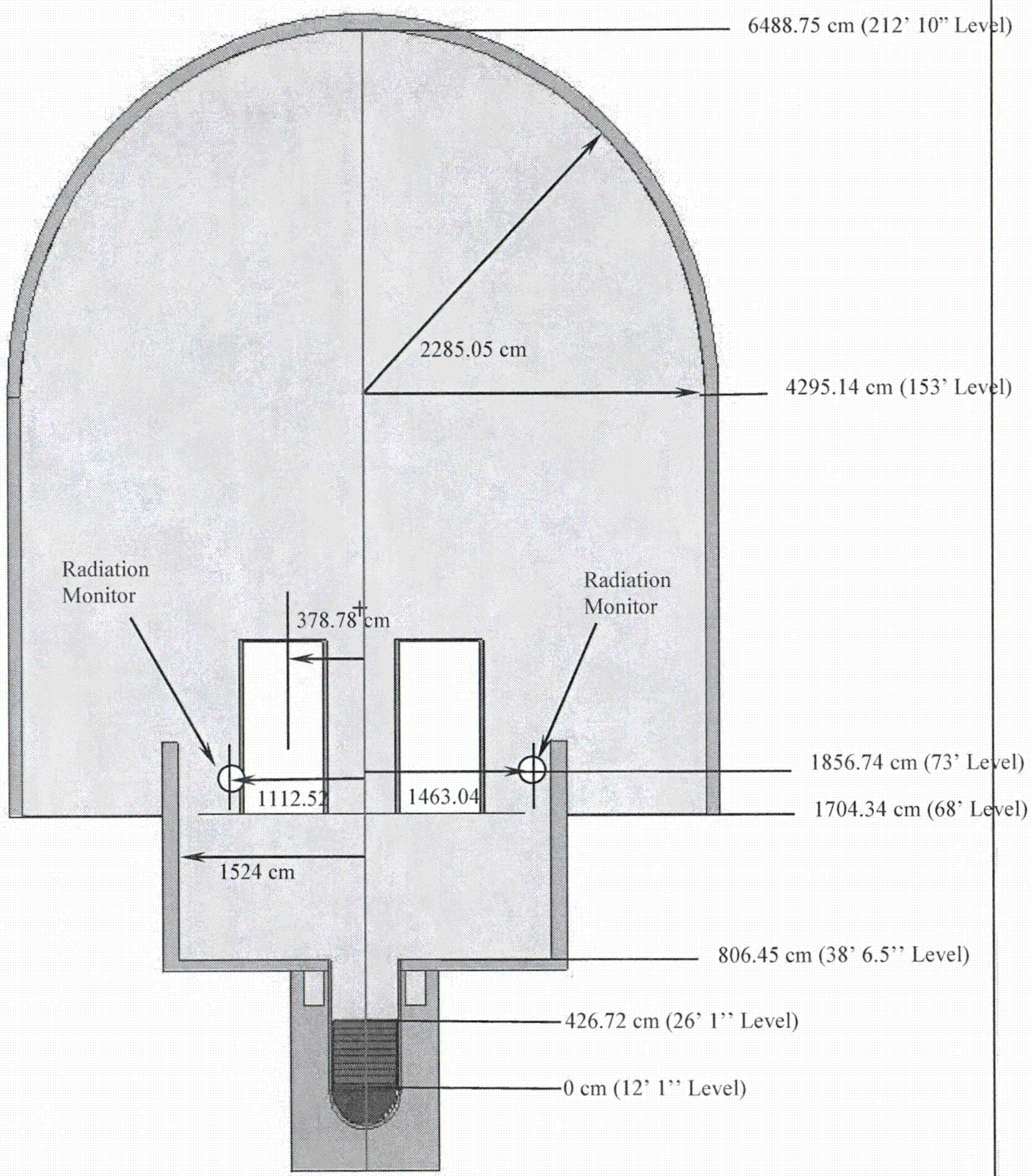


Figure 7-3 X-Z VISED Plot of Containment§



§ Modeled Steam Generators are not full height. Also, they are not on the same X-Z plane as the core shown above. They are included for visualization purposes.

Figure 7-4 Y-X VISED Plot of the Containment Geometry at Radiation Monitor Level

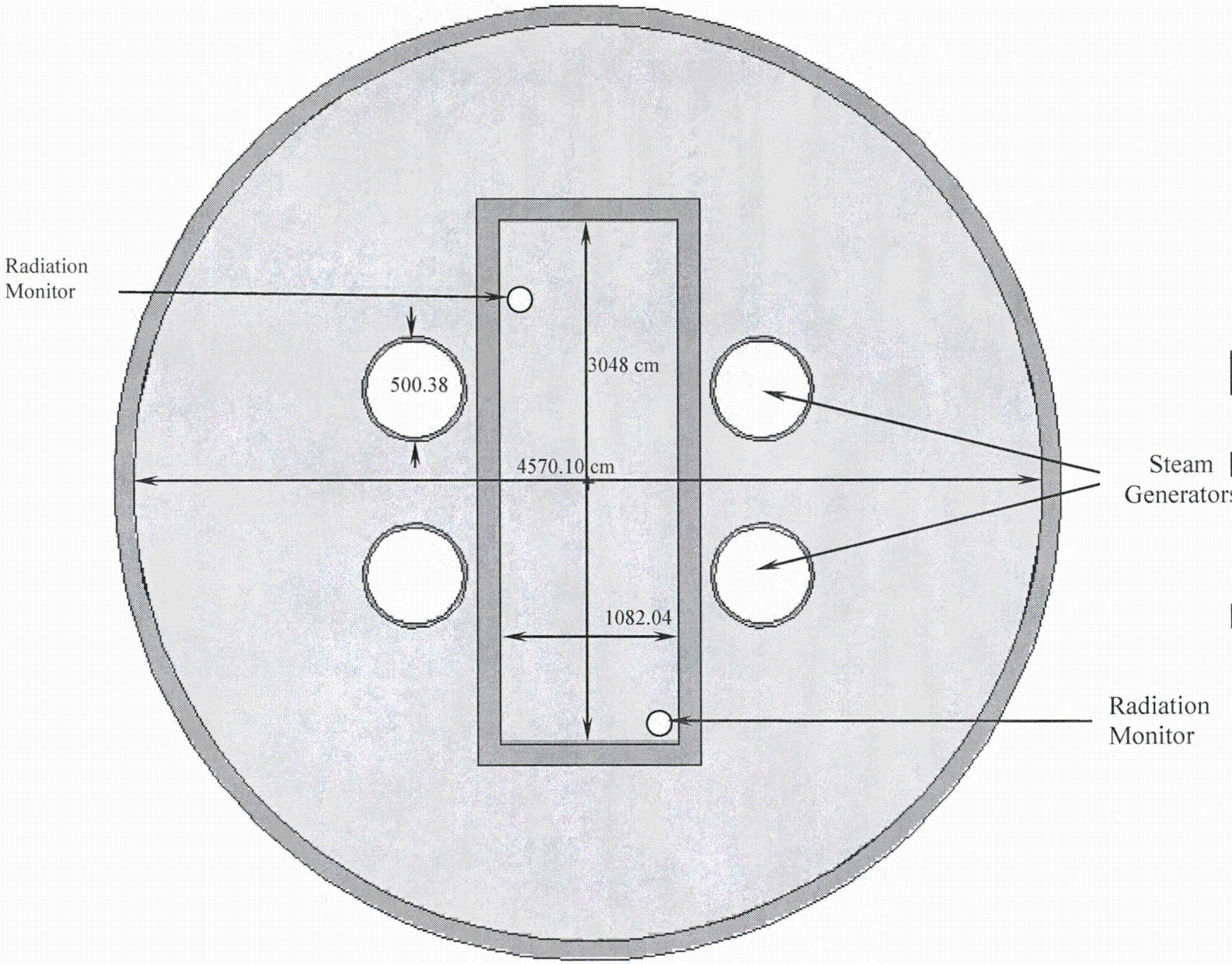


Table 7-2 Summary of Surfaces Used for MCNP Models

Surface Type	Surface Number	Dimensions							Description
RCC		X_0	Y_0	Z_0	V_X	V_Y	V_Z	R	
	1	0	0	0	0	0	426.72	193.675	Active Fuel Region
	2	0	0	0	0	0	806.45	219.71	Reactor Pressure Vessel Inner Surface
	3	0	0	0	0	0	806.45	245.11	Reactor Pressure Vessel Outer Surface
	30	0	0	426.72	0	0	60.96	219.71	0 to 2 ft Above Active Fuel Region
	31	0	0	806.45	0	0	18.26	245.11	Reactor Pressure Vessel Head
	32	0	0	487.68	0	0	60.96	219.71	0 to 2 ft Above Active Fuel Region (Applicable to Head Case Only)
	33	0	0	548.64	0	0	60.96	219.71	2 to 4 ft Above Active Fuel Region (Applicable to Head Case Only)

Surface Type	Surface Number	Dimensions							Description
RCC		X_0	Y_0	Z_0	V_X	V_Y	V_Z	R	
	34	0	0	609.60	0	0	60.96	219.71	4 to 6 ft Above Active Fuel Region (Applicable to Head Case Only)
	35	0	0	670.56	0	0	135.89	219.71	8 ft Above Active Fuel Region to RPV Head (Applicable to Head Case Only)
	41	0	0	525.78	0	0	213.36	275.59	Concrete Void for Primary Loop
	42	0	0	525.78	0	0	213.36	412.75	Concrete Void for Primary Loop
	10	0	0	739.14	0	0	67.31	245.11	Concrete Wall Cutout
	11	378.78	937.58	800.1	0	0	2057.08	250.19	Steam Generator 1
	12	378.78	937.58	820.1	0	0	2017.08	230.19	Steam Generator Inner 1

Surface Type	Surface Number	Dimensions							Description
	13	-378.78	937.58	800.1	0	0	2057.08	250.19	Steam Generator 2
	14	-378.78	937.58	820.1	0	0	2017.08	230.19	Steam Generator Inner 2
	15	-378.78	-937.58	800.1	0	0	2057.08	250.19	Steam Generator 3
	16	-378.78	-937.58	820.1	0	0	2017.08	230.19	Steam Generator Inner 3
	17	378.78	-937.58	800.1	0	0	2057.08	250.19	Steam Generator 4
	18	378.78	-937.58	820.1	0	0	2017.08	230.19	Steam Generator Inner 4
	21	0	0	1704.34	0	0	2590.8	2284.01	Containment Inner Liner Surface
	22	0	0	1704.34	0	0	2590.8	2285.05	Containment Inner Concrete Surface
	23	0	0	1704.34	0	0	2590.8	2376.49	Containment Outer Concrete Surface
RPP		**-X**	**X**	**-Y**	**Y**	**-Z**	**Z**		
	4	-497.84	497.84	-497.84	497.84	-497.84	739.14		Concrete Surrounding RPV

Surface Type	Surface Number	Dimensions							Description
RPP		-X	X	-Y	Y	-Z	Z		
	8	-1524	1524	-541.02	541.02	806.45	2161.54		Concrete Wall Fuel Pit Inner
	9	-1630.68	1630.68	-647.70	647.70	739.14	2161.54		Concrete Wall Fuel Pit Outer
SPH		X_0	Y_0	Z_0	R				
	5	0	0	0	219.71				Bottom of Reactor Pressure Vessel Inner
	6	0	0	0	245.11				Bottom of Reactor Pressure Vessel Outer
	24	0	0	4295.14	2284.10				Containment Dome Inner Liner Surface
	25	0	0	4295.14	2285.05				Containment Dome Inner Concrete Surface
	26	0	0	4295.14	2376.49				Containment Dome Outer Concrete Surface
PZ		Z							
	7	0							Fuel Bottom
	71	806.45							Top of RPV

Surface Type	Surface Number	Dimensions							Description
PZ		Z							
	20	426.72 (Variable)							Water Level (Variable for No Head Case Only)
	27	4295.14							Spring Line
	28	1704.34							68' Level
	101-110	42.672	-	426.72					Geometric Importance Divisions in Active Zone

Figure 7-5 MCNP Model Surface Cards[6]

```
c surfaces
1 rcc 0 0 0 0 0 426.72 193.675                    $ Active Fuel Region
2 rcc 0 0 0 0 0 806.45 219.71                     $ RPV Inner Surface
3 rcc 0 0 0 0 0 806.45 245.11                     $ RPV Outer Surface
31 rcc 0 0 806.45 0 0 18.26 245.11                $ Reactor Vessel Head
4 rpp -497.84 497.84 -497.84 497.84 -497.84 739.14  $ Concrete Surrounding RPV
41 rcc 0 0 525.78 0 0 213.36 275.59               $ Concrete Void for Primary Loop
42 rcc 0 0 525.78 0 0 213.36 412.75               $ Concrete Void for Primary Loop
5 sph 0 0 0 219.71                                $ Bottom of Reactor Pressure Vessel
6 sph 0 0 0 245.11                                $ Bottom of Reactor Pressure Vessel
7 pz 0                                            $ Bottom of Active Zone
71 pz 806.45                                      $ Top of RPV
8 rpp -1524 1524 -541.02 541.02 806.45 2161.54    $ Concrete Walls Fuel Pit Inner
9 rpp -1630.68 1630.68 -647.70 647.70 739.14 2161.54  $ Concrete Wall Fuel Pit Outer
10 rcc 0 0 739.14 0 0 67.31 245.11                $ Concrete Wall Cutout
11 rcc 378.78 937.58 800.1 0 0 2057.08 250.19     $ Steam Generator 1
12 rcc 378.78 937.58 820.1 0 0 2017.08 230.19     $ Inner Steam Generator 1
13 rcc -378.78 937.58 800.1 0 0 2057.08 250.19    $ Steam Generator 2
14 rcc -378.78 937.58 820.1 0 0 2017.08 230.19    $ Inner Steam Generator 2
15 rcc -378.78 -937.58 800.1 0 0 2057.08 250.19   $ Steam Generator 3
16 rcc -378.78 -937.58 820.1 0 0 2017.08 230.19   $ Inner Steam Generator 3
17 rcc 378.78 -937.58 800.1 0 0 2057.08 250.19    $ Steam Generator 4
18 rcc 378.78 -937.58 820.1 0 0 2017.08 230.19    $ Inner Steam Generator 4
20 pz 426.72                                      $ Water Elevation Surface
21 rcc 0 0 1704.34 0 0 2590.8 2284.01             $ Containment Inner Liner Surface
22 rcc 0 0 1704.34 0 0 2590.8 2285.05             $ Containment Inner Concrete Surface
23 rcc 0 0 1704.34 0 0 2590.8 2376.49             $ Containment Outer Concrete Surface
24 sph 0 0 4295.14 2284.01                        $ Containment Dome Inner Liner Surface
25 sph 0 0 4295.14 2285.05                        $ Ctmt. Dome Inner Concrete Surface
26 sph 0 0 4295.14 2376.49                        $ Ctmt. Dome Outer Concrete Surface
27 pz 4295.14                                     $ Spring Line
28 pz 1704.34                                     $ 68' Level
30 rcc 0 0 426.72 0 0 60.96 219.71                $ 0 to 2 ft above top of active fuel
32 rcc 0 0 487.68 0 0 60.96 219.71                $ 2 to 4 ft above top of active fuel
33 rcc 0 0 548.64 0 0 60.96 219.71                $ 4 to 6 ft above top of active fuel
34 rcc 0 0 609.60 0 0 60.96 219.71                $ 6 to 8 ft above top of active fuel
35 rcc 0 0 670.56 0 0 135.89 219.71               $ 8 ft above top of fuel to head
101 pz 42.672                                     $ Geometric Imp. Division Fuel Zone
102 pz 85.344                                     $ Geometric Imp. Division Fuel Zone
103 pz 128.016                                    $ Geometric Imp. Division Fuel Zone
104 pz 170.688                                    $ Geometric Imp. Division Fuel Zone
105 pz 213.36                                     $ Geometric Imp. Division Fuel Zone
106 pz 256.032                                    $ Geometric Imp. Division Fuel Zone
107 pz 298.704                                    $ Geometric Imp. Division Fuel Zone
108 pz 341.376                                    $ Geometric Imp. Division Fuel Zone
109 pz 384.048                                    $ Geometric Imp. Division Fuel Zone
110 pz 426.72                                     $ Geometric Imp. Division Fuel Zone
```

[6] The surface cards for the MCNP models without the reactor vessel head do not have surfaces 30, 31, 32, 33, 34 or 35. The other surfaces are identical, with the exception of surface 20 which is variable for the cases without the reactor vessel head only.

Figure 7-6 MCNP Model Cell Cards (No Head)

```
c cells
101 1 -4.38 -1 -101                     imp:p=1                 $ Active Fuel Region
102 1 -4.38 -1 101 -102                 imp:p=2                 $ Active Fuel Region
103 1 -4.38 -1 102 -103                 imp:p=3                 $ Active Fuel Region
104 1 -4.38 -1 103 -104                 imp:p=4                 $ Active Fuel Region
105 1 -4.38 -1 104 -105                 imp:p=8                 $ Active Fuel Region
106 1 -4.38 -1 105 -106                 imp:p=16                $ Active Fuel Region
107 1 -4.38 -1 106 -107                 imp:p=32                $ Active Fuel Region
108 1 -4.38 -1 107 -108                 imp:p=64                $ Active Fuel Region
109 1 -4.38 -1 108 -109                 imp:p=128               $ Active Fuel Region
110 1 -4.38 -1 109 -110                 imp:p=256               $ Active Fuel Region
2 2 -0.9982 1 -3 #4 -20                 imp:p=256               $ Water Region
4 4 -7.94 2 -3 7 -71                    imp:p=256               $ RPV Shell
5 4 -7.94 5 -6 -7 #7                    imp:p=256               $ Bottom RPV Shell
6 2 -0.9982 -5 -7                       imp:p=256               $ Water Above Fuel
61 2 -0.9982 -20 71 (-10:-8)            imp:p=256               $ Water Above Vessel Head
71 3 -1.21E-03 -42 41                   imp:p=256               $ Void for Primary Loop
7 5 -2.3 6 3 -4 #71                     imp:p=256               $ Concrete Surrounding RPV
8 5 -2.3 8 -9 10                        imp:p=256               $ Concrete above RPV
9 4 -7.94 -11 12 28                     imp:p=256               $ Steam Generator 1
10 0 -12 28                             imp:p=0                 $ Inner Steam Generator 1
11 4 -7.94 -13 14 28                    imp:p=256               $ Steam Generator 2
12 0 -14 28                             imp:p=0                 $ Inner Steam Generator 2
13 4 -7.94 -15 16 28                    imp:p=256               $ Steam Generator 3
14 0 -16 28                             imp:p=0                 $ Inner Steam Generator 3
15 4 -7.94 -17 18 28                    imp:p=256               $ Steam Generator 4
16 0 -18 28                             imp:p=0                 $ Inner Steam Generator 4
20 4 -7.94 21 -22                       imp:p=256               $ Containment Liner
21 5 -2.3 22 -23                        imp:p=256               $ Containment Wall
22 4 -7.94 24 -25 27                    imp:p=256               $ Containment Dome Liner
23 5 -2.3 25 -26 27                     imp:p=256               $ Containment Dome Concrete
24 5 -2.3 -21 -28 9 #21 #22 11 13
           15 17                        imp:p=256               $ 68 foot level
30 3 -1.21E-03 (-24:-21:-8:-10:-2)
          11 13 15 17 20 #8 #24 #2 1    imp:p=256               $ Air in Containment
999 0 1 #2 #4 #5 #6 #7 #71 #8 #9 #10
     #11 #12 #13 #14 #15 #16 #20 #21
     #22 #23 #24 #30 #61                imp:p=0                 $ Problem Boundary
```

Figure 7-7 X-Z VISED Plot of Reactor Vessel and Concrete Reactor Pit (With Head)

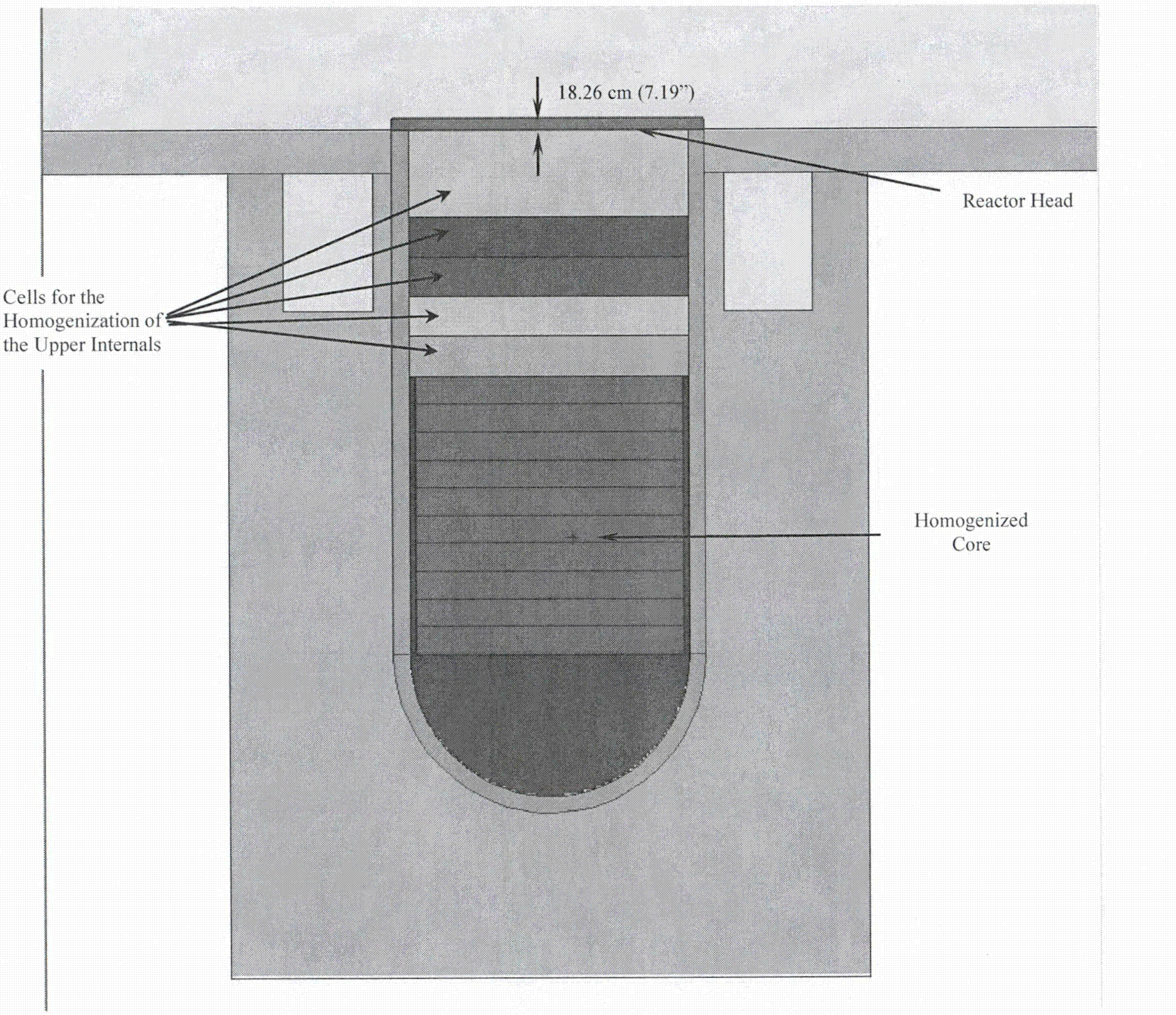


Figure 7-8 MCNP Cell Cards (With Head) [7]

```
c cells
101 1 -4.38  -1 -101              imp:p=1      $ Active Fuel Region (AFR)
102 1 -4.38  -1 101 -102          imp:p=2      $ Active Fuel Region
103 1 -4.38  -1 102 -103          imp:p=3      $ Active Fuel Region
104 1 -4.38  -1 103 -104          imp:p=4      $ Active Fuel Region
105 1 -4.38  -1 104 -105          imp:p=8      $ Active Fuel Region
106 1 -4.38  -1 105 -106          imp:p=16     $ Active Fuel Region
107 1 -4.38  -1 106 -107          imp:p=32     $ Active Fuel Region
108 1 -4.38  -1 107 -108          imp:p=64     $ Active Fuel Region
109 1 -4.38  -1 108 -109          imp:p=128    $ Active Fuel Region
110 1 -4.38  -1 109 -110          imp:p=256    $ Active Fuel Region
2 2 -0.9982 1 -3 #4 -20          imp:p=256    $ Water Region
41 4 -0.4177 -30                 imp:p=300    $ 0 to 2 feet above AFR
42 4 -1.2805 -32                 imp:p=400    $ 2 to 4 feet above AFR
43 4 -1.2805 -33                 imp:p=500    $ 4 to 6 feet above AFR
44 4 -1.2805 -34                 imp:p=600    $ 6 to 8 feet above AFR
45 4 -1.2805 -35                 imp:p=700    $ 8 feet above AFR
4 4 -7.94 2 -3 7 -71             imp:p=256    $ RPV Shell
5 4 -7.94 5 -6 -7 #7             imp:p=256    $ Bottom RPV Shell
6 2 -0.9982 -5 -7                imp:p=256    $ Water Above Fuel
62 6 -7.8212 -31                 imp:p=256    $ Reactor Vessel Head
61 2 -0.9982 -20 71 (-10:-8)     imp:p=256    $ Water Above Vessel Head
71 3 -1.21E-03 -42 41            imp:p=256    $ Void for Primary Loop
7 5 -2.3 6 3 -4 #71              imp:p=256    $ Concrete Surrounding RPV
8 5 -2.3 8 -9 3                  imp:p=256    $ Concrete above RPV
9 4 -7.94 -11 12 28              imp:p=256    $ Steam Generator 1
10 0 -12 28                      imp:p=0      $ Inner Steam Generator 1
11 4 -7.94 -13 14 28             imp:p=256    $ Steam Generator 2
12 0 -14 28                      imp:p=0      $ Inner Steam Generator 2
13 4 -7.94 -15 16 28             imp:p=256    $ Steam Generator 3
14 0 -16 28                      imp:p=0      $ Inner Steam Generator 3
15 4 -7.94 -17 18 28             imp:p=256    $ Steam Generator 4
16 0 -18 28                      imp:p=0      $ Inner Steam Generator 4
20 4 -7.94 21 -22                imp:p=256    $ Containment Liner
21 5 -2.3 22 -23                 imp:p=256    $ Containment Wall
22 4 -7.94 24 -25 27             imp:p=256    $ Containment Dome Liner
23 5 -2.3 25 -26 27              imp:p=256    $ Containment Dome Concrete
24 5 -2.3 -21 -28 9 #21 #22 11 13
         15 17                   imp:p=256    $ 68 foot level
30 3 -1.21E-03 (-24:-21:-8:) 11 13
        15 17 20 31 #8 #24 #2 1   imp:p=256    $ Air inside Containment
999 0 1 #2 #4 #5 #6 #7 #71 #8 #9 #10
     #11 #12 #13 #14 #15 #16 #20 #21
     #22 #23 #24 #30 #61 2 31     imp:p=0      $ External to Problem
```

[7] The material cards and densities for cells 41-44 are varied based on the modeled water level in the reactor vessel. Note that the carbon steel density is modeled with more significant digits than shown in the design input. This will not affect the calculated dose rates.

7.5 MCNP Source Definition

The core source term is assumed to be uniformly distributed throughout the volume, and has an energy spectra based on the core inventory [2]. Only the gamma source term is taken into account for this evaluation. Because the source term is generated shortly after shutdown, the fuel gamma source term will predominate. Therefore the N-gamma and hardware activation source terms can be neglected (Assumption 1). The source is defined on the MCNP *sdef* card using distributions to define the particle location and energy. The radius of the core is defined with the *rad* parameter, which automatically creates a uniform distribution based on a cylindrical geometry. The *ext* and *axs* parameters define the direction and distance of the cylinder axis. These parameters combined define the core where the particles can be born. The *erg* parameter defines the energy spectrum of source particles and is based on the results of the ORIGEN-S calculation discussed previously. This distribution is a histogram of energies represented by activities. These are automatically normalized by MCNP to create a probability distribution. The total activity is preserved in the tally multiplier. The MCNP source definition cards are shown below in Figure 7-9. The *sb* card is a source biasing card, which in this case biases the particle generation to the upper end of the core. This is a variance reduction technique to improve the statistical certainty in the results. Note that the core radius distribution is slightly larger than the diameter of the active fuel, which results in conservatively less self-shielding from the active fuel region.

Figure 7-9 MCNP Source Definition Cards

```
sdef rad=d1 ext=d2 axs=0 0 1 erg=d8                                    ←Source Definition Card
                                                                               -Radius = d1
                                                                               -Extent = d2
                                                                               -Axis = +Z
                                                                               -Energy = d8
si1 209.71                                                             ←Core Radius Distribution
si2 h 0 42.672 85.344 128.016 170.688 213.36 256.032 298.704           ←Core Axial Distribution
       341.376 384.048 426.72
sp2 0 1 1 1 1 1 1 1 1 1 1                                              ←Actual Uniform Distribution
sb2 0 0.001 0.001 0.01 0.01 0.01 0.1 0.1 0.1 1 1                       ←Biased to Top Distribution
c  Fuel Gamma Spectra
si8   h 1.000e-002 5.000e-002 1.000e-001 2.000e-001 3.000e-001 4.000e-001    ←Source Energy Groups
     6.000e-001 8.000e-001 1.000e+000 1.330e+000 1.660e+000 2.000e+000
     2.500e+000 3.000e+000 4.000e+000 5.000e+000 6.500e+000 8.000e+000
     1.000e+001 1.100e+001
sp8     0.00E+00 4.040E+19 1.306E+19 3.045E+19 1.893E+19 7.253E+18 1.511E+19 ←Source Emission on
                                                                                 Energy Basis
       2.766+19 4.457E+18 1.016E+18 7.389E+18 1.391E+17 1.558E+17 2.286E+17
       1.785E+15 3.827E+10 1.040E+09 1.741E+08 36950000 2001000
```

7.6 MCNP Tally Specification

The tallies used in this evaluation are point detectors placed at approximate locations of radiation monitors RE-8055 and RE-8099 for Unit 2, which approximates the Unit 1 configuration sufficiently such that only a single evaluation is performed. Point detectors are chosen because they use quasi-deterministic dose calculations that will provide better results than surface or cell based tallies that require the particles to enter those regions. The inputs to this card are the coordinates of the dose points followed by an exclusion zone (reduce variance), as well as a multiplier card, which represents the total core activity in photons/sec. The tally cards are shown in Figure 7-10. Note that the y dimensions are adjusted slightly from those in Table 5-2 to ensure no overlap between the detector and the modeled concrete wall.

Figure 7-10 MCNP Tally Cards

```
f5c RE-8055 and RE-8099                                   ←Tally Comment Card
f5:p -1112.52  -541.01 1856.7 20                          ←Tally 5 (point detector)
1463.04  541.01 1856.74 20                                x     y     z    exclusion
fm5 1.662E+20                                             ← Tally Multiplier
                                                            (Total Activity)
```

In addition, the flux is multiplied by ANSI/ANS flux-dose conversion factors [11]. This is specified in MCNP using the *de/df* cards. These are shown in Figure 7-11.

Figure 7-11 ANSI/ANS-6.1.1-1977 Gamma Flux to Dose Conversion Factors

```
c -----------------------------------------------------------------
c ANSI/ANS-6.1.1-1977
c Gamma Flux to Dose Conversion Factors
c (mrem/hr)/(photons/cm2-s)
c -----------------------------------------------------------------
de0  .01 .03 .05 .07 .10 .15 .20 .25 .30 .35 .40                       ←Energy Bins for Flux
     .45 .50 .55 .60 .65 .70 .80 1. 1.4 1.8 2.2                          to Dose Conversion
     2.6 2.8 3.25 3.75 4.25 4.75 5. 5.25 5.75 6.25
     6.75 7.5 9. 11.
df0  3.96E-03   5.82E-04   2.90E-04   2.58E-04   2.83E-04   3.79E-04      ←Energy Dependent
     5.01E-04   6.31E-04   7.59E-04   8.78E-04   9.85E-04   1.08E-03        Flux Multipliers
     1.17E-03   1.27E-03   1.36E-03   1.44E-03   1.52E-03   1.68E-03
     1.98E-03   2.51E-03   2.99E-03   3.42E-03   3.82E-03   4.01E-03
     4.41E-03   4.83E-03   5.23E-03   5.60E-03   5.80E-03   6.01E-03
     6.37E-03   6.74E-03   7.11E-03   7.66E-03   8.77E-03   1.03E-02
```

7.7 MCNP Material Cards

The MCNP material cards are provided in Figure 7-12. These are based on the compositions described in Table 5-4. Note that the compositions for air and stainless steel differ slightly from those in Table 5-4; these differences are minor and will not affect the calculated dose rates.

Figure 7-12 MCNP Material Cards[8]

```
m1     92235      -0.0256        $ Homogenized Active Fuel Region
       92238      -0.6146
       8016       -0.2056
       40000      -0.1367
       50000      -0.0020
       24000      -0.0001
       26000      -0.0003
       1001       -0.0149
m2 1001 2 8016 1                 $ Water
m3     6012 -0.000126            $ Air
       7014 -0.76508
       8016 -0.234793
m4     6000 -0.0008              $ SS 304
       14000 -0.01
       15031 -0.00045
       24000 -0.19
       25055 -0.02
       26000 -0.68375
       28000 -0.095
m5     26000 -0.014              $ Reg-Concrete
       1001 -0.01
       13027 -0.034
       20000 -0.044
       8016 -0.532
       14000 -0.337
       11023 -0.029
m6     6012 -0.01                 $ Carbon Steel
       26056 -0.99
m7     6000 -2.45E-04              $ Top Nozzle area with water
       14000 -3.06E-03
       15031 -1.38E-04
       24000 -5.82E-02
       25055 -6.13E-03
       26000 -2.09E-01
       28000 -2.91E-02
       8016 -6.17E-01
       1001 -7.71E-02
m8     6000 -4.84E-04              $ Upper Guide Structure with Water
       14000 -6.05E-03
       15031 -2.72E-04
       24000 -1.15E-01
       25055 -1.21E-02
       26000 -4.13E-01
       28000 -5.74E-02
       8016 -3.51E-01
       1001 -4.39E-02
```

[8] Material 7 is used in cell 40 depending on the water level of the core. Material 8 is used in cells 41, 42, and 43 depending on the water level of the core.

7.8 Results

File Naming Scheme:

The MCNP input files are named with the following convention:

P-height-condition where:

P = Project (STP)

Condition = h – with head
hi – with head, but mass of upper internals neglected
n – no head

Height = water height from top of active fuel region (ft)

Iteration (used for weight window optimization) = a-z

7.8.1 *Results without Head*

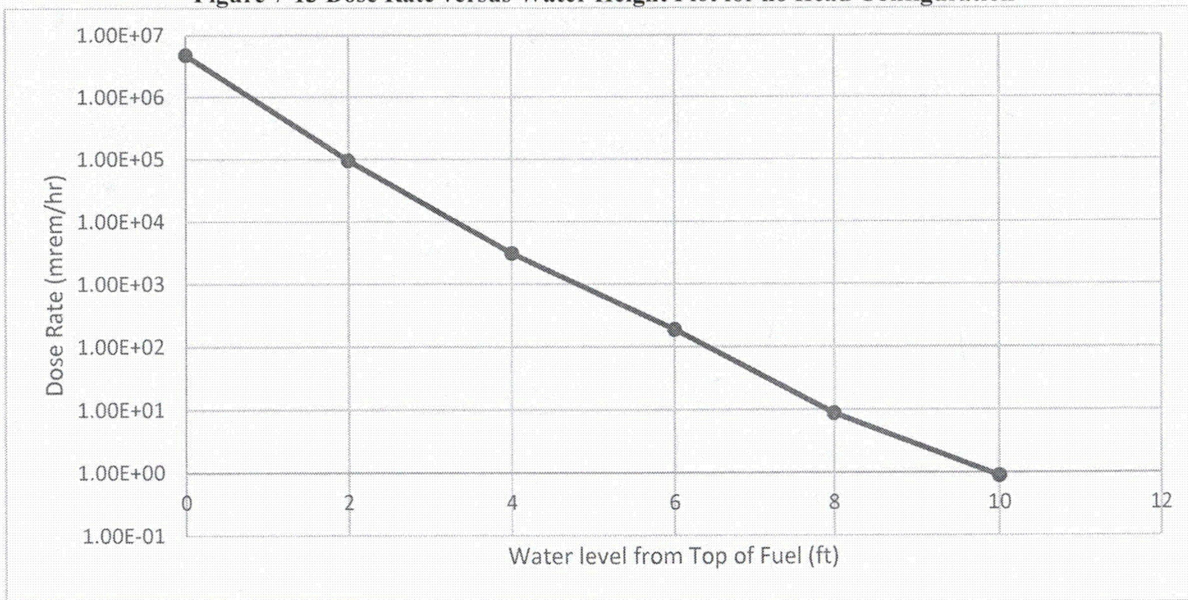
The dose rate as a function of water level is provided in Table 7-3 and the larger of the two dose rates is plotted in Figure 7-13, below. All of the water levels described in the following sections refer to the level at the top of the fuel (i.e. 0 foot water level is at the top of the fuel assemblies and ~12.5 feet is flange level).

Table 7-3 Dose Rate Response as a Function of Water Level for no Head Configuration (mrem/h)

Water Level (ft)	Dose Rate 1 RE-8055	fsd[9]	Dose Rate 2 RE-8099	fsd	Tally File
0	4.75E+06	4.52%	1.49E+06	4.40%	STPn0hm
2	9.41E+04	2.94%	3.51E+04	5.54%	STPn2hm
4	3.08E+03	6.76%	1.02E+03	8.73%	STPn4jm
6	1.88E+02	5.94%	8.28E+01	7.48%	STPn6nm
8	8.75E+00	7.14%	4.21E+00	12.47%	STPn8lm
10	8.83E-01	6.97%	4.07E-01	6.81%	STPn10rm

[9] Fraction standard deviation.

Figure 7-13 Dose Rate versus Water Height Plot for no Head Configuration



7.8.2 Results with Head

The dose rate results for the cases with the head in place are lower due to the lower ambient dose rate in the containment. The dose rates are listed in Table 7-4 and Table 7-5 and the higher of the two dose rates is plotted in Figure 7-14. Note that several uncertainties are higher than 10%, which is indicative of the difficulty in adequately converging a thick-shielded problem such as this. Because the slope of the curve is nearly linear, these results are judged to be adequate.

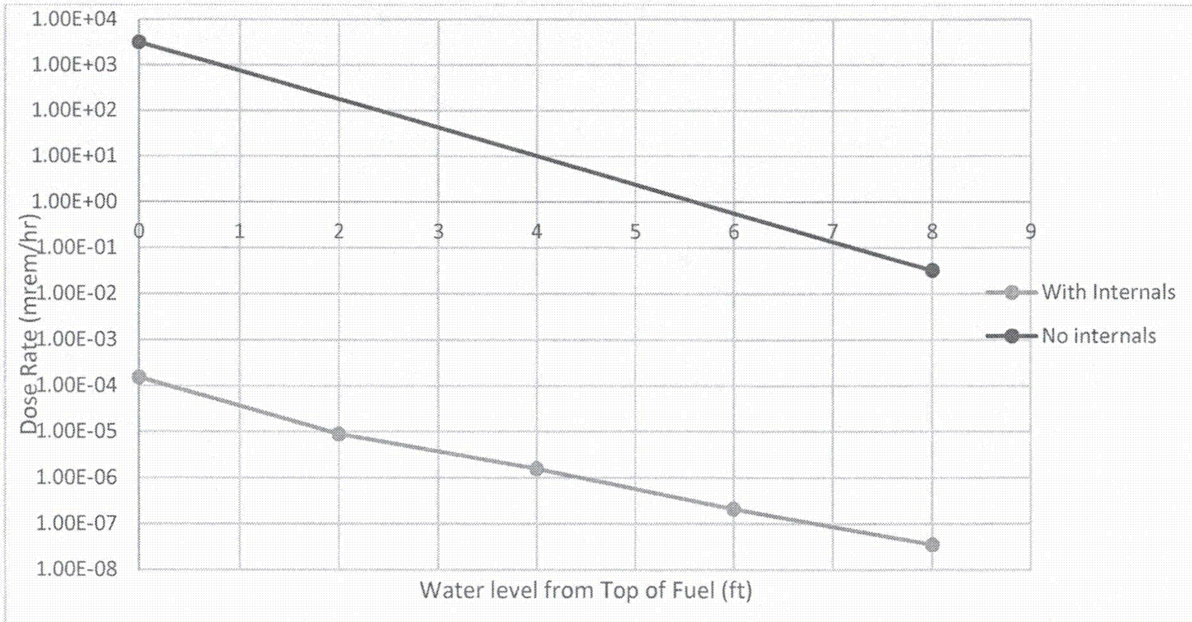
Table 7-4 Dose Rate Response as a Function of Water Level for Head on Configuration (mrem/h)

Water Level (ft)	Dose Rate 1 RE-8055	fsd	Dose Rate 2 RE-8099	fsd	Tally File
0	1.51E-04	10.03%	8.22E-05	11.04%	STPh0tm
2	8.98E-06	9.59%	4.73E-06	10.87%	STPh2qm
4	1.58E-06	13.41%	8.54E-07	11.53%	STPh4um
6	2.10E-07	24.79%	2.25E-07	31.82%	STPh6rm
8	3.50E-08	16.04%	8.35E-09	8.55%	STPh8um

Table 7-5 Dose Rate Response as a Function of Water Level for Head on Configuration neglecting Upper Internals and Upper Fuel Hardware mass (mrem/h)

Water Level (ft)	Dose Rate 1 RE-8055	fsd	Dose Rate 2 RE-8099	fsd	Tally File
0	3.16E+03	1.00%	1.50E+03	0.99%	STPhi0im
8	3.19E-02	10.86%	1.89E-02	16.23%	STPhi8mm

Figure 7-14 Dose Rate versus Water Height Plot for with Head Configuration



Appendix A – Electronic File Listing

```
Excel files:
06/18/2015  05:04 PM            21,632 STP Homogeneous.xlsx
06/17/2015  09:21 AM            17,783 STP.xlsx
06/19/2015  08:00 AM            37,096 STP_output.xlsx

Origen output:
06/08/2015  01:05 PM            86,560 STPEALa.out

MCNP output:

 Directory of \head\no internals\STPhi0
06/10/2015  03:02 AM         1,278,910 STPhi0io

 Directory of \head\no internals\STPhi8
06/18/2015  07:02 AM         2,258,187 STPhi8mo

 Directory of \head\STPh0
06/17/2015  07:53 AM         1,963,788 STPh0to

 Directory of \head\STPh2
06/18/2015  01:39 PM           344,948 STPh2qo

 Directory of \head\STPh4
06/19/2015  06:57 AM         4,469,733 STPh4uo

 Directory of \head\STPh6
06/19/2015  06:57 AM         1,312,757 STPh6ro

 Directory of \head\STPh8
06/19/2015  06:57 AM         1,280,159 STPh8uo

 Directory of \no head\STPn0
06/10/2015  07:16 PM         1,375,003 stpn0ho

 Directory of \no head\STPn2
06/10/2015  08:28 PM         1,275,392 STPn2ho

 Directory of \no head\STPn4
06/15/2015  03:13 AM         1,537,707 STPn4jo

 Directory of \no head\STPn6
06/19/2015  06:49 AM         2,894,751 STPn6no

 Directory of \no head\STPn8
06/18/2015  02:24 PM           269,152 STPn8lo

 Directory of \no head\STPn10
06/19/2015  07:14 AM         1,283,646 STPn10ro

Directory of \no head\sensitivity\STPnc0
06/26/2015  04:58 AM         1,712,210 stpnc0jo

Directory of \no head\sensitivity\STPnc2
06/26/2015  05:06 AM         1,725,224 stpnc2ao

Directory of \no head\sensitivity\STPnc4
06/26/2015  04:46 AM         1,608,957 stpnc4ao
```

Appendix B – Sensitivity Case Run

This appendix evaluates the impact of changes to the MCNP model for the No-Head, 0, 2 and 4 ft. water level case. The following changes to the model have been made for this case:

1. The dimension for concrete thickness on the side of the RPV is adjusted
2. The X and Y – Plane distances for the detectors are adjusted.
3. The model is updated to more accurately reflect the concrete at the 68' elevation in addition to the concrete "D-rings" around the Steam Generators.
4. The Upper Fuel Hardware region is included in the No-Head case as this hardware is connected to the fuel.

The Design Input Dimensions (Table 5-2) that have been updated by this case are as follows:

Dimension:	ft.	in	cm	reference
Reactor Pressure Vessel (RPV)				
Distance from Centerline to Edge of Concrete	18	6	563.88	[5]
Concrete Wall (D-ring)				
Inner Width	31	6	960.12	[7]
Inner Length	82	0	2499.36	[7]
Detector Locations				
Detector RE-8055 Distance from RPV (x-plane)	27	5	835.66	[6] and [19]
Detector RE-8055 Distance from RPV (y-plane)	15	3	464.82	[7] and [19]
Detector RE-8099 Distance from RPV (x-plane)	38	0	1158.24	[12] and [20]
Detector RE-8099 Distance from RPV (y-plane)	15	3	464.82	[7] and [20]

The Design Input Dimensions that have been added by this case are as follows:

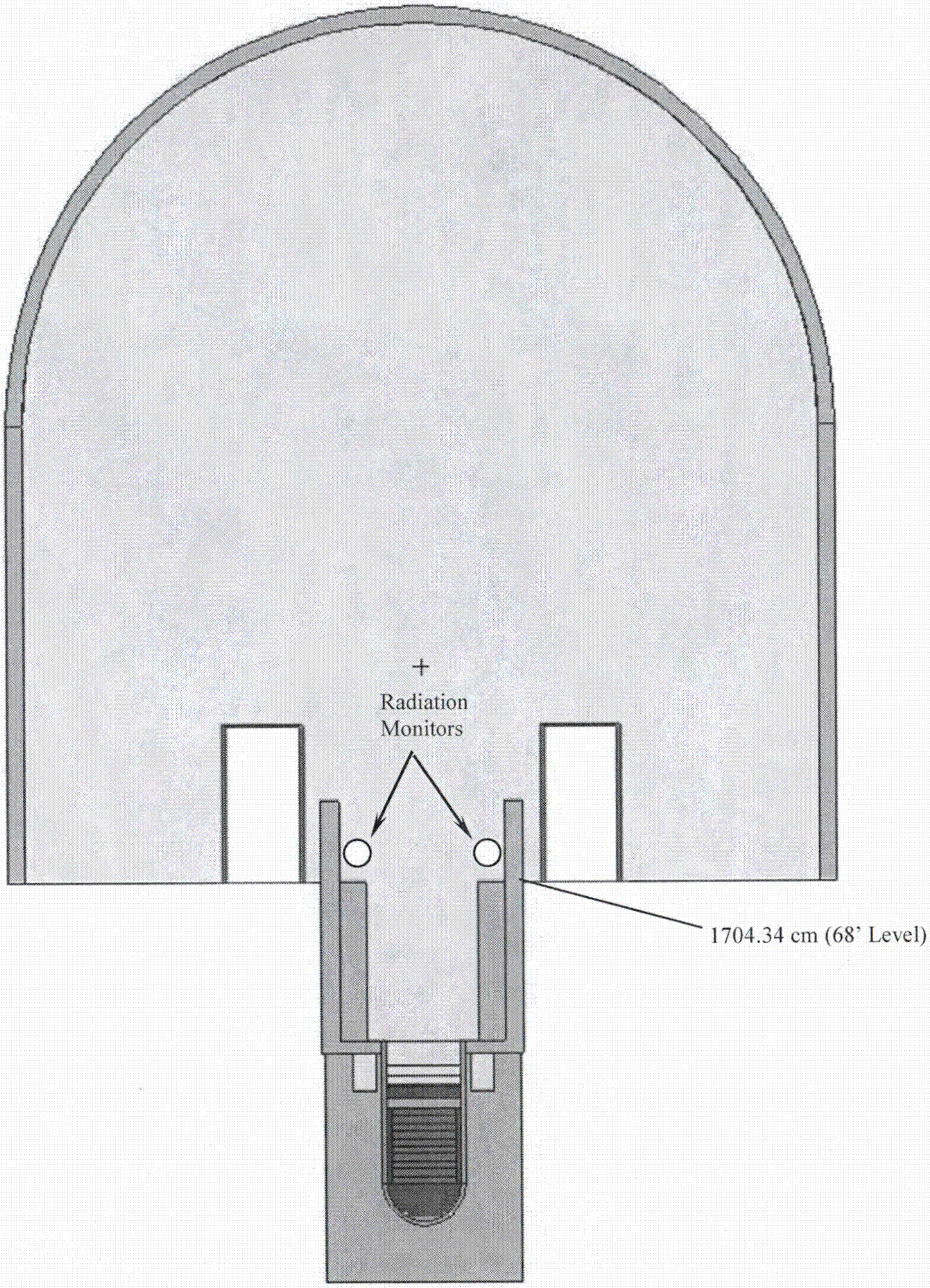
Dimension:	ft.	in	cm	reference
Concrete Wall (68' Elevation)				
Lower Elevation	26	5 1/2	806.45	[9]
Upper Elevation	55	11	1704.34	[7]
Overall Height			897.89	Calculated
Thickness	5	3	160.02	[6] and [7]

Dimension:	ft.	in	cm	reference
Inner Width	21	0	640.08	[6]
Inner Length	71	6	2499.36	[6] and [7]*

*Modeled 5' 3" from the edge of the "D ring" wall.

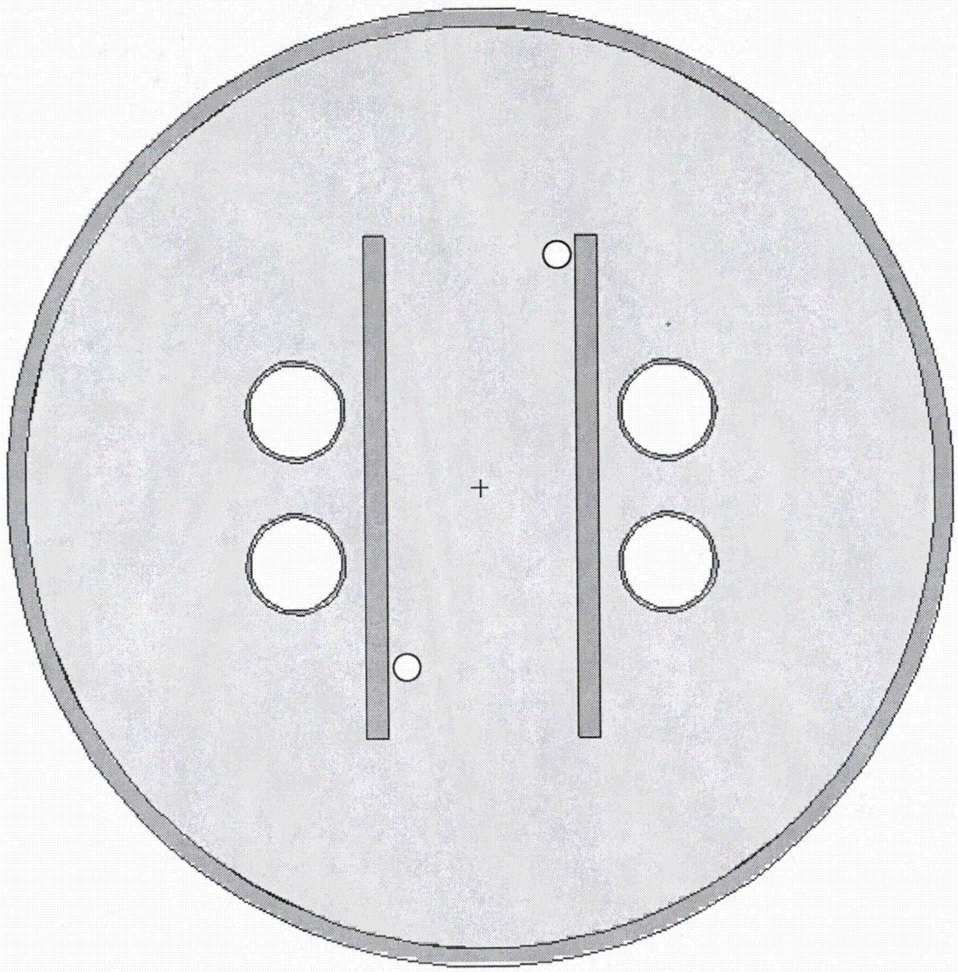
The sensitivity case includes modeling of the Upper Fuel Hardware for the No-Head case, therefore the cells for the Upper Internals Homogenization are utilized. The cell 41 and 42 material cards are modified as necessary to reflect the water level in the core. Cells 43-45 are modeled as air.

Y-Z VISED Plot of Containment[10]



[10] Modeled Steam Generators are not full height. Also, they are not on the same Y-Z plane as the core shown above. They are included for visualization purposes.

Y-X VISED Plot of the Containment Geometry at Radiation Monitor Level[11]



[11] Note that the north and south "D ring" walls were removed from the model by extending cell 8 along the x axis.

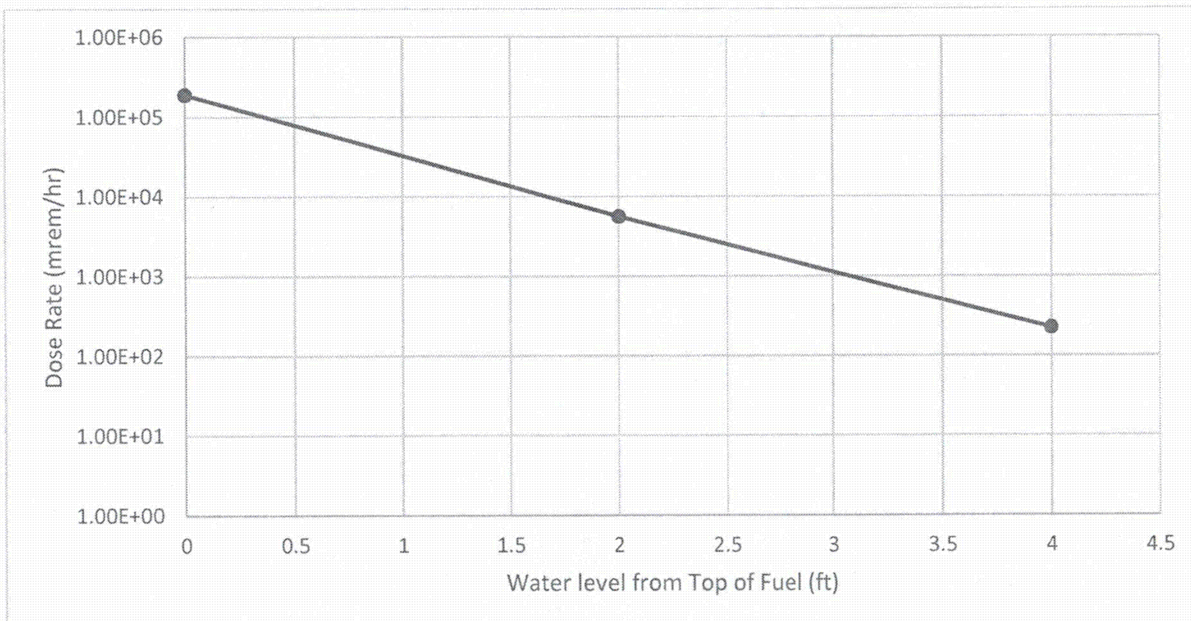
The surfaces that were added or updated for the model are shown below:

Summary of Surfaces Added or Updated for Sensitivity Case

Surface Type	Surface Number	Dimensions						Description
RPP		-X	X	-Y	Y	-Z	Z	
	4	-563.88	563.88	-563.88	563.88	-563.88	739.14	Concrete Surrounding RPV
	8	-1249.68	1249.68	-480.06	480.06	806.45	2161.54	Concrete Walls "D ring" Inner
	9	-1249.68	1249.68	-586.74	586.74	739.14	2161.54	Concrete Walls "D ring" Outer
	81	-1089.66	1089.66	-320.04	320.04	806.45	1704.34	Concrete to 68' Elevation Inner
	91	-1249.68	1249.68	-480.06	480.06	806.45	1704.34	Concrete to 68' Elevation Outer

The dose rates for the No Head, 0, 2 and 4 ft. water level cases with the changes described above are shown in the table and graph below.

Water Level (ft)	Dose Rate 1 RE-8055	fsd[§§§]	Dose Rate 2 RE-8099	fsd	Tally File
0	3.00E+05	1.37%	1.89E+05	2.64%	STPnc0jm
2	9.35E+03	3.58%	5.64E+03	3.85%	STPnc2am
4	3.78E+02	10.16%	2.25E+02	8.74%	STPnc4am



[§§§] Fractional standard deviation.