PROCEDURE COVER SHEET

SUSQUEHANNA, LLC PROCEDURE	
EAL CLASSIFICATION BASES ADHERENCE LEVEL: INFORMATION USE	02/13/2015 EP-RM-004 Revision [X] Page 1 of 281

QUALITY CLASSIFICATION:	APPROVAL CLASSIFICATION:
(X) QA Program ( ) Non-QA Program	(X) Plant ( ) Non-Plant ( ) Instruction

EFFECTIVE DATE: ____________

PERIODIC REVIEW FREQUENCY: 2 Year

PERIODIC REVIEW DUE DATE: REWL X0200

RECOMMENDED REVIEWS: All

Procedure Owner: Emergency Planning

Responsible Supervisor: Manager-EP

Responsible FUM: Manager-EP

Responsible Approver: Manager-NRA

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CAUTION

This Reference Manual shall follow the process described in NDAP-QA-0004, Procedure Change Process, for subsequent revisions AND NOT the change process described under EP-112, Emergency Plan Reference Manual Program. It must be maintained in accordance with 10 CFR50.54(q).

1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Susquehanna, LLC. It should be used to facilitate review of the Susquehanna EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EP-PS-100, Emergency Director Control Room (ref. 4.2.1), EP-PS-101, TSC Emergency Director (ref. 4.2.2) and EP-PS-200, Recovery Manager (ref. 4.2.3), may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Director/Recovery Manager in making classifications, particularly those involving judgment or multiple events. The bases information may also be useful in training and for explaining event classifications to off-site officials.

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

2.0 DISCUSSION

2.1 Background

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the Susquehanna Emergency Plan.

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

NEI 99-01 (NUMARC/NESP-007) Revisions 4 and 5 were subsequently issued for industry implementation. Enhancements over earlier revisions included:

- Consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur during plant shutdown conditions.
- Initiating conditions and example emergency action levels that fully address conditions that may be postulated to occur at permanently Defueled Stations and Independent Spent Fuel Storage Installations (ISFSIs).
- Simplifying the fission product barrier EAL threshold for a Site Area Emergency.

Subsequently, Revision 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 (ADAMS Accession Number ML110240324) (ref.

4.1.1), Susquehanna conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

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.

Many of the EALs derived from the NEI methodology are fission product barrier threshold based. That is, the conditions that define the EALs are based upon thresholds that represent the loss or potential loss of one or more of the three fission product barriers. "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. A "Loss" threshold means the barrier no longer assures containment of radioactive materials. A "Potential Loss" threshold implies an increased probability of barrier loss and decreased certainty of maintaining the barrier.

The primary fission product barriers are:

A. <u>Fuel Clad (FC):</u> The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.

B. <u>Reactor Coolant System (RCS):</u> The RCS Barrier is the reactor coolant system pressure boundary and includes the Reactor Pressure Vessel (RPV) and all reactor coolant system piping up to and including the isolation valves.

C. <u>Primary Containment (PC):</u> The Primary Containment Barrier includes the drywell, the wetwell, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. Primary Containment Barrier thresholds are used as criteria for escalation of the Emergency Classification Level (ECL) from Alert to a Site Area Emergency or a General Emergency.

2.3 Fission Product Barrier Classification Criteria

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

<u>Alert:</u>

Any loss or any potential loss of either Fuel Clad or RCS barrier

<u>Site Area Emergency:</u>

Loss or potential loss of any two barriers

<u>General Emergency:</u>

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

2.4 EAL Organization

The Susquehanna EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
  - EALs applicable under <u>all</u> plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
  - EALs applicable only under <u>hot</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Startup, or Power Operation mode.
  - EALs applicable only under <u>cold</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling or Defueled mode.

  The purpose of the groups is to avoid review of hot condition EALs when the plant is in a cold condition and avoid review of cold condition EALs when the plant is in a hot condition. This approach significantly minimizes the total number of EALs that must be reviewed by the EAL-user for a given plant condition, reduces EAL-user reading burden and, thereby, speeds identification of the EAL that applies to the emergency.

- Within each group, assignment of EALs to categories and subcategories:

  Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. The Susquehanna EAL categories are aligned to and represent the NEI 99-01"Recognition Categories." Subcategories are used in the Susquehanna scheme as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The Susquehanna EAL categories and subcategories are listed in Table 2.4-1.

  The primary tool for determining the emergency classification level is the EAL Classification Matrix. The bases provide the EAL user with the background and justification behind the EAL threshold values identified using the guidance set forth in NEI 99-01 Revision 6. If there is any doubt with regard to the applicability of any EAL, the technical basis should be reviewed. The user should consult Section 3.0 and Attachments 1 (EAL technical bases) & 2 (fission product barrier technical bases) of this document for such information.

Table 2.4-1 EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
All Operating Mode:	
R – Abnormal **R**ad Levels / Rad Effluent	1 – Radiological Effluent 2 – Irradiated Fuel Event 3 – Area Radiation Levels
H – **H**azards and Other Conditions Affecting Plant Safety	1 – Security 2 – Seismic Event 3 – Natural or Technological Hazard 4 – Fire 5 – Hazardous Gas 6 – Control Room Evacuation 7 – ED/RM Judgment
E – Independent Spent Fuel Storage Installation (ISFSI)	1 – Confinement Boundary
Hot Conditions:	
S – **S**ystem Malfunction	1 – Loss of Essential AC Power 2 – Loss of Vital DC Power 3 – Loss of Control Room Indications 4 – RCS Activity 5 – RCS Leakage 6 – RPS Failure 7 – Loss of Communications 8 – Hazardous Event Affecting Safety Systems
F – Fission Product Barrier Degradation	None
Cold Conditions:	
C – **C**old Shutdown / Refueling System Malfunction	1 – RPV Level 2 – Loss of Essential AC Power 3 – Loss of Vital DC Power 4 – RCS Temperature 5 – Loss of Communications 6 – Hazardous Event Affecting Safety Systems

2.5 Technical Bases Information

EAL technical bases are provided in Attachment 1 for each EAL according to EAL group (All, Hot, Cold), EAL category (R, C, H, S, F and E) and EAL subcategory. A summary explanation of each category and subcategory is given at the beginning of the technical bases discussions of the EALs included in the category. For each EAL, the following information is provided:

<u>Category Letter & Title</u>

<u>Subcategory Number & Title</u>

<u>Initiating Condition (IC)</u>

Site-specific description of the generic IC given in NEI 99-01 Rev. 6.

<u>EAL Identifier (enclosed in rectangle)</u>

Each EAL is assigned a unique identifier to support accurate communication of the emergency classification to onsite and offsite personnel. Four characters define each EAL identifier:

1. First character (letter): Corresponds to the EAL category as described above (R, C, H, S, F or E)
2. Second character (letter): The emergency classification (G, S, A or U)

   G = General Emergency
   S = Site Area Emergency
   A = Alert
   U = Unusual Event
3. Third character (number): Subcategory number within the given category. Subcategories are sequentially numbered beginning with the number one (1). If a category does not have a subcategory, this character is assigned the number one (1).
4. Fourth character (number): The numerical sequence of the EAL within the EAL subcategory. If the subcategory has only one EAL, it is given the number one (1).

<u>Classification (enclosed in rectangle):</u>

Unusual Event (U), Alert (A), Site Area Emergency (S) or General Emergency (G)

<u>EAL (enclosed in rectangle)</u>

Exact wording of the EAL as it appears in the EAL Classification Matrix

<u>Mode Applicability</u>

One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Power Operations, 2 - Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refueling, D - Defueled, or All. (See Section 2.6 for operating mode definitions)

<u>Definitions:</u>

If the EAL wording contains a defined term, the definition of the term is included in this section. These definitions can also be found in Section 5.1. Defined terms used in Initiating Condition and Emergency Action Level statements are set in all capital letters (e.g., ALL CAPS).

Basis:

A Plant-Specific basis section that provides Susquehanna-relevant information concerning the EAL. This is followed by a Generic basis section that provides a description of the rationale for the EAL as provided in NEI 99-01 Rev. 6.

Susquehanna Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.6 Operating Mode Applicability (ref. 4.1.2 except Defueled)

1 Power Operation
Reactor is critical and the mode switch is in RUN

2 Startup
The mode switch is in REFUEL (with all reactor vessel head closure bolts fully tensioned) or STARTUP/HOT STANDBY

3 Hot Shutdown
The mode switch is in SHUTDOWN (with one or more reactor vessel head closure bolts less than fully tensioned) and average reactor coolant temperature is >200°F

4 Cold Shutdown
The mode switch is in SHUTDOWN (with one or more reactor vessel head closure bolts less than fully tensioned) and average reactor coolant temperature is ≤ 200°F

5 Refueling
The mode switch is in REFUEL or SHUTDOWN with one or more reactor vessel head closure bolts are less than fully tensioned

D Defueled
All fuel removed from the reactor vessel (i.e., full core offload during refueling or extended outage) (ref. 4.1.1)

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

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Emergency Director/Recovery Manager 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 based on loss or potential loss of Fission Product Barrier Thresholds.

3.1.1 Classification Timeliness

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" (ref. 4.1.3).

3.1.2 Valid Indications

All emergency classification assessments shall 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, verification could be accomplished through an instrument channel check, response on related or redundant indicators, or direct observation by plant personnel.

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

3.1.3 Imminent Conditions

For ICs and EALs that have a stipulated time duration (e.g., 15 minutes, 30 minutes, etc.), the Emergency Director/Recovery Manager 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.

3.1.4 Planned vs. Unplanned Events

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 (ref. 4.1.4).

3.1.5 Classification Based on Analysis

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.). For these EALs, the EAL wording 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).

3.1.6 Emergency Director/Recovery Manager Judgment

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. The NEI 99-01 EAL scheme provides the Emergency Director/Recovery Manager 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/Recovery Manager 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 in the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

3.2 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 must be consistent with the related Operating Mode Applicability and Notes. If an EAL has been met or exceeded, the associated IC is likewise met, the emergency classification process “clock” starts, and the ECL must be declared in accordance with plant procedures no later than fifteen minutes after the process “clock” started.

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 (ref. 4.1.3).

3.2.1 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 both 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 both 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* (ref. 4.1.5).

3.2.2 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.

3.2.3 Classification of Imminent Conditions

Although EALs provide specific thresholds, the Emergency Director/Recovery Manager 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/Recovery Manager, 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.

3.2.4 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.

As noted above, guidance concerning classification of rapidly escalating events or conditions is provided in RIS 2007-02 (ref. 4.1.5).

3.2.5 Classification of Short-Lived Events

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 an earthquake or a failure of the reactor protection system to automatically scram the reactor followed by a successful manual scram.

3.2.6 Classification of Transient Conditions

Many of the ICs and/or EALs 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.

EAL momentarily met during expected plant response - 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.

EAL momentarily met but the condition is corrected prior to an emergency declaration – 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 high pressure ECCS systems fail to automatically start. RPV level rapidly decreases and the plant enters an inadequate core cooling condition (a potential loss of both the fuel clad and RCS barriers). If an operator manually starts a high pressure ECCS system in accordance with an EOP step and clears the inadequate core cooling 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 (process clock) 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 when an operator is able to take a successful corrective action prior to the Emergency Director/Recovery Manager 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.

3.2.7 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 (ref. 4.1.6) is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR § 50.72 (ref. 4.1.4) 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.

3.2.8 Retraction of an Emergency Declaration

Guidance on the retraction of an emergency declaration reported to the NRC is discussed in NUREG-1022 (ref. 4.1.6).

4.0 REFERENCES

4.1 Developmental

4.1.1 NEI 99-01 Revision 6, Methodology for the Development of Emergency Action Levels for Non-Passive Reactors, ADAMS Accession Number ML110240324

4.1.2 Technical Specifications Table 1.1-1 Modes

4.1.3 NSIR/DPR-ISG-01 Interim Staff Guidance, Emergency Planning for Nuclear Power Plants

4.1.4 10 § CFR 50.72 Immediate Notification Requirements for Operating Nuclear Power Reactors

4.1.5 RIS 2007-02 Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events, February 2, 2007

4.1.6 NUREG-1022 Event Reporting Guidelines: 10CFR50.72 and 50.73

4.1.7 NUH-003 Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Section 1.3.1

4.1.8 NDAP-QA-0309 Primary Containment Access and Control

4.1.9 NDAP-QA-0321 Secondary Containment Integrity Control

4.1.10 Susquehanna LLC, Susquehanna Steam Electric Station Emergency Plan, Section 1.0, Definitions

4.1.11 10 § CFR 50.73 License Event Report System

4.1.12 ON-FPC-101(201) Loss of Fuel Pool Cooling

4.1.13 EO-000-104 Secondary Containment Control

4.1.14 FSAR Section 3.7a Seismic Design

4.2 Implementing

4.2.1 EP-PS-100 Emergency Director Control Room

4.2.2 EP-PS-101 TSC Emergency Director

4.2.3 EP-PS-200 Recovery Manager

4.2.4 NEI 99-01 Rev. 6 to Susquehanna EAL Comparison Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

Selected defined terms used in Initiating Condition and 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.*

CAN/CANNOT BE MAINTAINED ABOVE/BELOW

The value of an identified parameter is/is not able to be held within the specified limit. The determination requires an evaluation of system performance and availability in relation to parameter values and trends. An instruction prescribing action when a parameter cannot be *maintained above or below a specified limit neither requires nor prohibits anticipatory action-* depending upon plant conditions, the action may be taken as soon as it is determined that the limit will ultimately be exceeded, or delayed until the limit is actually reached. Once the parameter does exceed the limit, however, the action must be performed; it may not be delayed while attempts are made to restore the parameter to within the desired control band.

CAN/CANNOT BE RESTORED ABOVE/BELOW

The value of an identified parameter is/is not able to be brought within the specified limit. The determination requires an evaluation of system performance and availability in relation to parameter values and trends. An instruction prescribing action when a value CANNOT BE RESTORED AND MAINTAINED above or below a specified limit does not require immediate action simply because the current values is outside the range, but does not permit extended operation beyond the limit; the action must be taken as soon as it is apparent that the specified range cannot be attained.

CONFINEMENT BOUNDARY

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the Susquehanna ISFSI, Confinement Boundary is defined as the Dry Shielded Canister (DSC) (Ref. 4.1.7).

CONTAINMENT CLOSURE

The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to Susquehanna, CONTAINMENT CLOSURE is established per NDAP-QA-0309 (ref 4.1.8) for Primary Containment OR is established per NDAP-QA-0321 (ref 4.1.9) for Secondary Containment.

EMERGENCY PLAN BOUNDARY (EPB) (ref. 4.1.10)

Same as the Exclusion Area Boundary, i.e., that area around SSES within a radius of 1800 feet determined in accordance with 10 CFR 100.11.

EPA PAGs

Environment Protection Agency Protective Action Guidelines. The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs require Susquehanna to recommend protective actions for the general public to offsite planning agencies.

The dose program complies with the "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," (EPA-400), adopting the dose calculation methodology I ICRP #26/30. The accident dose assessments are based on the adult physiology per EPA 400, except for one case – that is, child thyroid dose conversion factors are used in calculating thyroid CDE. However adult physiology is used in calculating thyroid CDE for purposes of evaluating the need for a sheltering only PAR and evaluating controlled venting of containment. Calculations of TEDE are made using the (adult) dose factors provided in EPA-400.

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 require a post-event inspection to determine if the attributes of an explosion are present.

FIRE

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

FLOODING

A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

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 Susquehanna 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 Susquehanna. 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.

IMPEDE(D)

Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective measures such as temporary shielding, SCBAs or dose extensions beyond Emergency Plan RWP that are not routinely employed to access the room/area).

INTRUSION

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

MAINTAIN

Take appropriate action to hold the value of an identified parameter within specified limits.

NORMAL LEVELS

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

OPERATING BASIS EARTHQUAKE (OBE)

An earthquake which, considering the regional and local geology, and seismology and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant. It is that earthquake which produces the vibratory ground motion for which these features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional (ref. 4.1.14).

OWNER CONTROLLED AREA (ref. 4.1.10)

Includes the area within the expanded security perimeter, i.e., the areas that are bordered by the Vehicle Barriers System. The OWNER CONTROLLED AREA also includes the Monitored OWNER CONTROLLED AREA (MOCA) as defined in Security Procedures.

PROJECTILE

An object directed toward a Nuclear Power Plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA (ref. 4.1.10)

Area within the station inner security fence (PROTECTED AREA Barrier) designated to implement the requirements of 10 CFR 73.

RCS INTACT

The RCS should be considered intact when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals). **REFUELING**

PATHWAY

The reactor refueling cavity, spent fuel pool and fuel transfer canal comprise the refueling pathway (ref. 4.1.12).

RESTORE

Take the appropriate action required to return the value of an identified parameter to the applicable limits

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 (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

SAFE SHUTDOWN EARTHQUAKE (SSE)

An earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology, and seismology and specific characteristics of local subsurface material. It is that earthquake which produces the maximum vibratory ground motion for which Seismic Category I systems and components are designed to remain functional (ref 4.1.14).

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.

SITE BOUNDARY

The line beyond which the land is not owned, leased or otherwise controlled by the licensee (Susquehanna drawing C243786, Sh 1, "Site Facilities and Boundary Map.") (ref. 4.1.10).

UNISOLABLE

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

- The term UNISOLABLE also includes any decision by plant staff or procedure direction to not isolate a primary system.
- Normal leakage past a closed isolation valve is not considered UNISOLABLE leakage.

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.

VALID

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

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.

5.2 Abbreviations/Acronyms

°F .... Degrees Fahrenheit
° .... Degrees
AC .... Alternating Current
AOP .... Abnormal Operating Procedure
APRM .... Average Power Range Meter
ATWS .... *Anticipated Transient Without Scram*
BWR .... Boiling Water Reactor
BWROG .... Boiling Water Reactor Owners Group
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
CS .... Core Spray
DBA .... Design Basis Accident
DC .... Direct Current
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
ED .... Emergency Director
EDST .... Evaporator Distillate Sample Tank
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
EPB .... Emergency Plan Boundary
EPG .... Emergency Procedure Guideline
EPIP .... Emergency Plan Implementing Procedure
ESF .... Engineered Safety Feature
ESS .... Engineered Safeguards System
FAA .... Federal Aviation Administration
FBI .... Federal Bureau of Investigation
FEMA .... Federal Emergency Management Agency
FPRR .... Fire Protection Review Report
FSAR .... Final Safety Analysis Report
GE .... General Emergency
HCTL .... Heat Capacity Temperature Limit
HPCI .... High Pressure Coolant Injection
IC .... Initiating Condition
IPEEE .... Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI .... Independent Spent Fuel Storage Installation
K_{eff} .... Effective Neutron Multiplication Factor
LCO .... Limiting Condition of Operation

LER .... Licensee Event Report
LOCA .... Loss of Coolant Accident
LRW .... Liquid Radwaste
LRW .... Light Water Reactor
MPC .... Maximum Permissible Concentration/Multi-Purpose Canister
MPH .... Miles Per Hour
MSIV .... Main Steam Isolation Valve
MSL .... Main Steam Line
mR, mRem, mrem, mREM .... milli-Roentgen Equivalent Man
MW .... Megawatt
NEI .... Nuclear Energy Institute
NESP .... National Environmental Studies Project
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
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM/ODAM .... Off-site Dose Calculation (Assessment) Manual
ORO .... Offsite Response Organization
PA .... Protected Area
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR .... Pressurized Water Reactor
PSIG .... Pounds per Square Inch Gauge
R .... Roentgen
RB .... Reactor Building
RCIC .... Reactor Core Isolation Cooling
RCS .... Reactor Coolant System
Rem, rem, REM .... Roentgen Equivalent Man
RETS .... Radiological Effluent Technical Specifications
RHR .... Residual Heat Removal
RM .... Recovery Manager
RPS .... Reactor Protection System
RPV .... Reactor Pressure Vessel
RRC .... Reactor Recirculation
RWCU .... Reactor Water Cleanup
SAR .... Safety Analysis Report
SBO .... Station Blackout
SCBA .... Self-Contained Breathing Apparatus

SDHR .... Supplemental Decay Heat Removal
SGTS .... Stand-By Gas Treatment System
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
SSE .... Safe Shutdown Earthquake
SSES .... Susquehanna Steam Electric Station
SW .... Service Water
TEDE .... Total Effective Dose Equivalent
TAF .... Top of Active Fuel
TRM .... Technical Requirements Manual
TSC .... Technical Support Center

6.0 Susquehanna-TO-NEI 99-01 Rev. 6 EAL CROSS-REFERENCE

This cross-reference is provided to facilitate association and location of a Susquehanna EAL within the NEI 99-01 IC/EAL identification scheme. Further information regarding the development of the Susquehanna EALs based on the NEI guidance can be found in the EAL Comparison Matrix.

Susquehanna EAL	NEI 99-01 Rev. 6	
	IC	Example EAL
RU1.1	AU1	1, 2
RU1.2	AU1	3
RU2.1	AU2	1
RA1.1	AA1	1
RA1.2	AA1	2
RA1.3	AA1	3
RA1.4	AA1	4
RA2.1	AA2	1
RA2.2	AA2	2
RA2.3	AA2	3
RA3.1	AA3	1
RA3.2	AA3	2
RS1.1	AS1	1
RS1.2	AS1	2
RS1.3	AS1	3
RS2.1	AS2	1
RG1.1	AG1	1
RG1.2	AG1	2
RG1.3	AG1	3
RG2.1	AG2	1
CU1.1	CU1	1

Susquehanna EAL	NEI 99-01 Rev. 6	
	IC	Example EAL
CU1.2	CU1	2
CU2.1	CU2	1
CU3.1	CU4	1
CU4.1	CU3	1
CU4.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1, 2, 3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA4.1	CA3	1, 2
CA6.1	CA6	1
CS1.1	CS1	1
CS1.2	CS1	2
CS1.3	CS1	3
CG1.1	CG1	1
CG1.2	CG1	2
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1, 2 3
HU2.1	HU2	1
HU3.1	HU3	1
HU3.2	HU3	2
HU3.3	HU3	3

Susquehanna EAL	NEI 99-01 Rev. 6	
	IC	Example EAL
HU3.4	HU3	4
N/A	HU3	5
HU4.1	HU4	1
HU4.2	HU4	2
HU4.3	HU4	3
HU4.4	HU4	4
HU7.1	HU7	1
HA1.1	HA1	1, 2
HA5.1	HA5	1
HA6.1	HA6	1
HA7.1	HA7	1
HS1.1	HS1	1
HS6.1	HS6	1
HS7.1	HS7	1
HG1.1	HG1	1
HG7.1	HG7	1
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	1
SU4.2	SU3	2
SU5.1	SU4	1, 2, 3
SU6.1	SU5	1
SU6.2	SU5	2
SU7.1	SU6	1, 2, 3
SA1.1	SA1	1

Susquehanna EAL	NEI 99-01 Rev. 6	
	IC	Example EAL
SA3.1	SA2	1
SA6.1	SA5	1
SA8.1	SA9	1
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1
SG1.1	SG1	1
SG1.2	SG8	1
EU1.1	E-HU1	1

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Fission Product Barrier Matrix and Bases

7.3 Attachment 3, Safe Shutdown Room/Areas Tables R-2 & H-2 Bases

7.4 Attachment 4, Table R – Abnormal Rad Levels / Rad Effluents (Form EP-RM-004-R)

7.5 Attachment 5, Table E – ISFSI (Form EP-RM-004-E)

7.6 Attachment 6, Table H – Hazards (Form EP-RM-004-H)

7.7 Attachment 7, Table S – System Malfunctions (Form EP-RM-004-S)

7.8 Attachment 8, Table F – Fission Product Barrier Degradation (Form EP-RM-004-F)

7.9 Attachment 9, Table C – Cold Shutdown/Refueling System Malfunctions (Form EP-RM-004-C)

ATTACHMENT 1

Emergency Action Level Technical Bases

Category R – Abnormal Rad Release / Rad Effluent

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

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

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

Events of this category pertain to the following subcategories:

1. Radiological Effluent

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

2. Irradiated Fuel Event

   Conditions indicative of a loss of adequate shielding or damage to irradiated fuel may preclude access to vital plant areas or result in radiological releases that warrant emergency classification.

3. Area Radiation Levels

   Sustained general area radiation levels which may preclude access to areas requiring continuous occupancy also warrant emergency classification.

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.1	**General Emergency**
Gaseous effluent > Table R-1 column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4)	

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Effluent Monitor Classification Thresholds (Note 4)

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Plant Vent (noble gas)	0C630 0C677	1.9E+09 µCi/min (site total)	1.9E+08 µCi/min (site total)	1.9E+07 µCi/min (site total)	4.0E+06 µCi/min (site total)
Liquid	LRW	RR-06433	----	----	----	2 x hi alarm
Liquid	1(2) RHRSW A/B	RR-D12-1(2)R606	----	----	----	2 x hi alarm
Liquid	1(2) SW/SDHR	RR-D12-1(2)R606	----	----	----	2 x hi alarm

Mode Applicability:

All

Definition(s):

None

Susquehanna Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to EMERGENCY PLAN BOUNDARY doses that exceed either (ref. 1):

- 1000 mrem TEDE
- *5000 mrem CDE Child Thyroid*

The SITE BOUNDARY is irregularly shaped and therefore would result in dose projections and protective action recommendations that can vary significantly depending on plume direction and affected sector. Using dose projections calculated using the EPB (i.e. EXCLUSION AREA) provides a more consistent approach to Public Protective Action Recommendations since the EPB is more consistently defined in all directions. The EPB is at or within the SITE BOUNDARY in all compass sectors.

The EPA PAGs are expressed in terms of the projected sum of the effective dose equivalent (EDE) from external radiation and the committed effective dose equivalent (CEDE) incurred from inhalation of radioactive materials, or as the committed dose equivalent (CDE) to the thyroid. For the purpose of these IC/EALs, the projected dose quantity Total Effective Dose Equivalent (TEDE), as defined in 10 CFR 20 is used in lieu of "the projected sum of...EDE...and CEDE...", with CEDE considering significant dose from inhaled radionuclides during the early phase of the event. The EPA protective action guidance provides for the use of adult thyroid dose conversion factors. However, the Commonwealth of Pennsylvania requires the use of child thyroid CDE for purposes of comparison of projected thyroid CDE to the PAG for thyroid CDE.

The monitor reading threshold for RG1.1 was determined as described in ref. 1.

The column "GE" gaseous effluent release values in Table R-1 correspond to calculated doses of 100% of the EPA Protective Action Guidelines (TEDE or CDE child Thyroid). For multi-release point gaseous releases, classification should be based on dose assessment that considers the total site release rate.

The SPING *monitors the radioactive effluent from the Units 1 and 2 Turbine Building and* Reactor Building Ventilation Stacks and the Standby Gas Treatment System Exhaust Vent. All five collectively are the Plant Vent on Table R-1. The SPING system is normally aligned to be operated from Brother Control Terminals 0C630 in the Control Room, using 0C677 as a backup *in the TSC. Three Post-Accident Vent Stack Sampling Systems* (PAVSSS) have been installed as backup to the SPING Units. They are used following an accident involving fuel degradation if the SPING monitoring capabilities are lost. Control Terminal CT-1 with System Operator Console CT-1B interrogates each of the SPING and PAVSSS Radiation Monitors for particulate, iodine, and noble gases and informs the operator of changes in operational status, alarm condition, or system parameters within seconds of their occurrence. (ref. 2)

NEI 99-01Basis:

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.

Susquehanna Basis Reference(s):

1. NEP Technical Basis 02-005 Rev. #2 Noble Gas Release Rate Limits for EALs
2. FSAR 18.1.30 Accident-Monitoring Instrumentation
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.2	**General Emergency**
Dose assessment using actual meteorology indicates doses > 1,000 mrem TEDE or 5,000 mrem child thyroid CDE at or beyond the EMERGENCY PLAN BOUNDARY (Notes 3, 4)	

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

EMERGENCY PLAN BOUNDARY (EPB) - Same as the Exclusion Area Boundary, i.e., that area around SSES within a radius of 1800 feet determined in accordance with 10 CFR 100.11

Susquehanna Basis:

Dose assessments are performed by computer-based methods (ref. 2).

The EMERGENCY PLAN BOUNDARY (EPB) is used in assessing dose effects to the public rather than the SITE BOUNDARY. The EPB is at or within the SITE BOUNDARY in all compass sectors. The SSES dose projection model (MIDAS) utilizes the EPB when performing dose calculations (ref. 5).

The SITE BOUNDARY is irregularly shaped and therefore would result in dose projections and protective action recommendations that can vary significantly depending on plume direction and affected sector. Using dose projections calculated using the EPB (i.e. Exclusion Area) provides a more consistent approach to Public Protective Action Recommendations since the EPB is more consistently defined in all directions. The EPB is at or within the SITE BOUNDARY in all compass sectors.

The EPA PAGs are expressed in terms of the projected sum of the effective dose equivalent (EDE) from external radiation and the committed effective dose equivalent (CEDE) incurred from inhalation of radioactive materials, or as the committed dose equivalent (CDE) to the thyroid. For the purpose of these IC/EALs, the projected dose quantity Total Effective Dose Equivalent (TEDE), as defined in 10 CFR 20 is used in lieu of "the projected sum of...EDE...and CEDE...", with CEDE considering significant dose from inhaled radionuclides during the early phase of the event. The EPA protective action guidance provides for the use of adult thyroid dose conversion factors. However, the Commonwealth of Pennsylvania requires the use of child thyroid CDE for purposes of comparison of projected thyroid CDE to the PAG for thyroid CDE.

Since dose assessment is based on actual meteorology, whereas the monitor reading RG1.1 is not, the results from these assessments may indicate that the classification is not warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading RG1.1. Classification should not be delayed pending the results of these dose assessments.

NEI 99-01 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.

Susquehanna Basis Reference(s):

1. Susquehanna LLC, Susquehanna Steam Electric Stations Emergency Plan, Section 7.1.1, Off Site Dose Calculations
2. EP-RM-005 SSES MIDAS-NU User Manual
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.3 General Emergency

Field survey results indicate **EITHER** of the following at or beyond the EMERGENCY PLAN BOUNDARY:

- Closed window dose rates > 1,000 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate child thyroid CDE > 5,000 mrem for 60 min. of inhalation.

(Notes 1, 2)

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

EMERGENCY PLAN BOUNDARY (EPB) - Same as the Exclusion Area Boundary, i.e., that area around SSES within a radius of 1800 feet determined in accordance with 10 CFR 100.11

Susquehanna Basis:

The EMERGENCY PLAN BOUNDARY (EPB) is used in assessing dose effects to the public rather than the SITE BOUNDARY. The EPB is at or within the SITE BOUNDARY in all compass sectors. The SSES dose projection model (MIDAS) utilizes the EPB when performing dose calculations (ref. 1).

The SITE BOUNDARY is irregularly shaped and therefore would result in dose projections and protective action recommendations that can vary significantly depending on plume direction and affected sector. Using dose projections calculated using the EPB (i.e. EXCLUSION AREA) provides a more consistent approach to Public Protective Action Recommendations since the EPB is more consistently defined in all directions. The EPB is at or within the SITE BOUNDARY in all compass sectors.

The EPA PAGs are expressed in terms of the projected sum of the effective dose equivalent (EDE) from external radiation and the committed effective dose equivalent (CEDE) incurred from inhalation of radioactive materials, or as the committed dose equivalent (CDE) to the thyroid. For the purpose of these IC/EALs, the projected dose quantity Total Effective Dose

Equivalent (TEDE), as defined in 10 CFR 20 is used in lieu of "the projected sum of...EDE...and CEDE...", with CEDE considering significant dose from inhaled radionuclides during the early phase of the event. The EPA protective action guidance provides for the use of adult thyroid dose conversion factors. However, the Commonwealth of Pennsylvania requires the use of child thyroid CDE for purposes of comparison of projected thyroid CDE to the PAG for thyroid CDE.

NEI 99-01 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.

Susquehanna Basis Reference(s):

1. EP-RM-005 SSES MIDAS-NU User Manual
2. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.1 Site Area Emergency

Gaseous effluent > Table R-1 column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4)

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Effluent Monitor Classification Thresholds (Note 4)

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Plant Vent (noble gas)	0C630 0C677	1.9E+09 µCi/min (site total)	1.9E+08 µCi/min (site total)	1.9E+07 µCi/min (site total)	4.0E+06 µCi/min (site total)
Liquid	LRW	RR-06433	----	----	----	2 x hi alarm
Liquid	1(2) RHRSW A/B	RR-D12-1(2)R606	----	----	----	2 x hi alarm
Liquid	1(2) SW/SDHR	RR-D12-1(2)R606	----	----	----	2 x hi alarm

Mode Applicability:

All

Definition(s):

None

Susquehanna Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to EMERGENCY PLAN BOUNDARY (EPB) doses that exceed either (ref. 1):

- 100 mrem TEDE
- 500 mrem CDE Child Thyroid

The SITE BOUNDARY is irregularly shaped and therefore would result in dose projections and protective action recommendations that can vary significantly depending on plume direction and affected sector. Using dose projections calculated using the EPB (i.e. EXCLUSION AREA) provides a more consistent approach to Public Protective Action Recommendations since the EPB is more consistently defined in all directions. The EPB is at or within the SITE BOUNDARY in all compass sectors.

The EPA PAGs are expressed in terms of the projected sum of the effective dose equivalent (EDE) from external radiation and the committed effective dose equivalent (CEDE) incurred from inhalation of radioactive materials, or as the committed dose equivalent (CDE) to the thyroid. For the purpose of these IC/EALs, the projected dose quantity Total Effective Dose Equivalent (TEDE), as defined in 10 CFR 20 is used in lieu of "the projected sum of...EDE...and CEDE...", with CEDE considering significant dose from inhaled radionuclides during the early phase of the event. The EPA protective action guidance provides for the use of adult thyroid dose conversion factors. However, the Commonwealth of Pennsylvania requires the use of child thyroid CDE for purposes of comparison of projected thyroid CDE to the PAG for thyroid CDE.

The monitor reading threshold for RS1.1 was determined as described in ref. 1.

The column "SAE" gaseous effluent release values in Table R-1 correspond to calculated doses of 10% of the EPA Protective Action Guidelines (TEDE or CDE child Thyroid). For multi-release point gaseous releases, classification should be based on dose assessment that considers the total site release rate.

The SPING monitors the radioactive effluent from the Units 1 and 2 Turbine Building and Reactor Building Ventilation Stacks and the Standby Gas Treatment System Exhaust Vent. All five collectively are the Plant Vent on Table R-1. The SPING system is normally aligned to be operated from Brother Control Terminals 0C630 in the Control Room, using 0C677 as a backup in the TSC. Three Post-Accident Vent Stack Sampling Systems (PAVSSS) have been installed as backup to the SPING Units. They are used following an accident involving fuel degradation if the SPING monitoring capabilities are lost. Control Terminal CT-1 with System Operator Console CT-1B interrogates each of the SPING and PAVSSS Radiation Monitors for particulate, iodine, and noble gases and informs the operator of changes in operational status, alarm condition, or system parameters within seconds of their occurrence. (ref. 2)

NEI 99-01 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.

Susquehanna Basis Reference(s):

1. NEP Technical Basis 02-005 Rev. #2 Noble Gas Release Rate Limits for EALs
2. FSAR 18.1.30 Accident-Monitoring Instrumentation
3. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.2	**Site Area Emergency**
Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem child thyroid CDE at or beyond the EMERGENCY PLAN BOUNDARY (Notes 3, 4)	

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

EMERGENCY PLAN BOUNDARY (EPB) - Same as the Exclusion Area Boundary, i.e., that area around SSES within a radius of 1800 feet determined in accordance with 10 CFR 100.11

Susquehanna Basis:

Dose assessments are performed by computer-based methods (ref. 2).

The EMERGENCY PLAN BOUNDARY (EPB) is used in assessing dose effects to the public rather than the SITE BOUNDARY. The EPB is at or within the SITE BOUNDARY in all compass sectors. The SSES dose projection model (MIDAS) utilizes the EPB when performing dose calculations (ref. 5).

The SITE BOUNDARY is irregularly shaped and therefore would result in dose projections and protective action recommendations that can vary significantly depending on plume direction and affected sector. Using dose projections calculated using the EPB (i.e. Exclusion Area) provides a more consistent approach to Public Protective Action Recommendations since the EPB is more consistently defined in all directions. The EPB is at or within the SITE BOUNDARY in all compass sectors.

The EPA PAGs are expressed in terms of the projected sum of the effective dose equivalent (EDE) from external radiation and the committed effective dose equivalent (CEDE) incurred from inhalation of radioactive materials, or as the committed dose equivalent (CDE) to the thyroid. For the purpose of these IC/EALs, the projected dose quantity Total Effective Dose Equivalent (TEDE), as defined in 10 CFR 20 is used in lieu of "the projected sum of...EDE...and CEDE...", with CEDE considering significant dose from inhaled radionuclides during the early phase of the event. The EPA protective action guidance provides for the use of adult thyroid dose conversion factors. However, the Commonwealth of Pennsylvania requires the use of child thyroid CDE for purposes of comparison of projected thyroid CDE to the PAG for thyroid CDE.

Since dose assessment is based on actual meteorology, whereas the monitor reading RS1.1 is not, the results from these assessments may indicate that the classification is not warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading RS1.1. Classification should not be delayed pending the results of these dose assessments.

NEI 99-01 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.

Susquehanna Basis Reference(s):

1. Susquehanna LLC, Susquehanna Steam Electric Stations Emergency Plan, Section 7.1.1, Off Site Dose Calculations
2. EP-RM-005 SSES MIDAS-NU User Manual
3. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.3 Site Area Emergency

Field survey results indicate **EITHER** of the following at or beyond the EMERGENCY PLAN BOUNDARY:

- Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate child thyroid CDE > 500 mrem for 60 min. of inhalation.

(Notes 1, 2)

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

EMERGENCY PLAN BOUNDARY (EPB) - Same as the Exclusion Area Boundary, i.e., that area around SSES within a radius of 1800 feet determined in accordance with 10 CFR 100.11.

Susquehanna Basis:

The EMERGENCY PLAN BOUNDARY (EPB) is used in assessing dose effects to the public rather than the SITE BOUNDARY. The EPB is at or within the SITE BOUNDARY in all compass sectors. The SSES dose projection model (MIDAS) utilizes the EPB when performing dose calculations (ref. 1).

The SITE BOUNDARY is irregularly shaped and therefore would result in dose projections and protective action recommendations that can vary significantly depending on plume direction and affected sector. Using dose projections calculated using the EPB (i.e. EXCLUSION AREA) provides a more consistent approach to Public Protective Action Recommendations since the EPB is more consistently defined in all directions. The EPB is at or within the SITE BOUNDARY in all compass sectors

The EPA PAGs are expressed in terms of the projected sum of the effective dose equivalent (EDE) from external radiation and the committed effective dose equivalent (CEDE) incurred from inhalation of radioactive materials, or as the committed dose equivalent (CDE) to the thyroid. For the purpose of these IC/EALs, the projected dose quantity Total Effective Dose Equivalent (TEDE), as defined in 10 CFR 20 is used in lieu of "the projected sum of...EDE...and

CEDE...", with CEDE considering significant dose from inhaled radionuclides during the early phase of the event. The EPA protective action guidance provides for the use of adult thyroid dose conversion factors. However, the Commonwealth of Pennsylvania requires the use of child thyroid CDE for purposes of comparison of projected thyroid CDE to the PAG for thyroid CDE.

Since dose assessment is based on actual meteorology, whereas the monitor reading RS1.1 is not, the results from these assessments may indicate that the classification is not warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading RS1.1. Classification should not be delayed pending the results of these dose assessments.

NEI 99-01Basis:

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.

Escalation of the emergency classification level would be via IC RG1.

Susquehanna Basis Reference(s):

1. EP-RM-005 SSES MIDAS-NU User Manual
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.1	**Alert**
Gaseous effluent > Table R-1 column "Alert" for ≥ 15 min. (Notes 1, 2, 3, 4)	

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4 The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Effluent Monitor Classification Thresholds (Note 4)

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Plant Vent (noble gas)	0C630 0C677	1.9E+09 µCi/min (site total)	1.9E+08 µCi/min (site total)	1.9E+07 µCi/min (site total)	4.0E+06 µCi/min (site total)
Liquid	LRW	RR-06433	----	----	----	2 x hi alarm
Liquid	1(2) RHRSW A/B	RR-D12-1(2)R606	----	----	----	2 x hi alarm
Liquid	1(2) SW/SDHR	RR-D12-1(2)R604	----	----	----	2 x hi alarm

Mode Applicability:

All

Definition(s):

None

Susquehanna Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to EMERGENCY PLAN BOUNDARY (EPB) doses that exceed either (ref. 1):

- 10 mrem TEDE
- 50 mrem CDE Child Thyroid

The SITE BOUNDARY is irregularly shaped and therefore would result in dose projections and protective action recommendations that can vary significantly depending on plume direction and affected sector. Using dose projections calculated using the EPB (i.e. Exclusion Area) provides a more consistent approach to Public Protective Action Recommendations since the EPB is more consistently defined in all directions. The EPB is at or within the SITE BOUNDARY in all compass sectors.

The EPA PAGs are expressed in terms of the projected sum of the effective dose equivalent (EDE) from external radiation and the committed effective dose equivalent (CEDE) incurred from inhalation of radioactive materials, or as the committed dose equivalent (CDE) to the thyroid. For the purpose of these IC/EALs, the projected dose quantity Total Effective Dose Equivalent (TEDE), as defined in 10 CFR 20 is used in lieu of "the projected sum of...EDE...and CEDE...", with CEDE considering significant dose from inhaled radionuclides during the early phase of the event. The EPA protective action guidance provides for the use of adult thyroid dose conversion factors. However, the Commonwealth of Pennsylvania requires the use of child thyroid CDE for purposes of comparison of projected thyroid CDE to the PAG for thyroid CDE.

The monitor reading threshold for RA1.1 was determined as described in ref. 1.

The column "ALERT" gaseous effluent release value in Table R-1 correspond to calculated doses of 1% (10% of the SAE thresholds) of the EPA Protective Action Guidelines (TEDE or CDE Thyroid). For multi-release point gaseous releases, classification should be based on dose assessment that considers the total site release rate.

The SPING monitors the radioactive effluent from the Units 1 and 2 Turbine Building and Reactor Building Ventilation Stacks and the Standby Gas Treatment System Exhaust Vent. All five collectively are the Plant Vent on Table R-1. The SPING system is normally aligned to be operated from Brother Control Terminals 0C630 in the Control Room, using 0C677 as a backup in the TSC. Three Post-Accident Vent Stack Sampling Systems (PAVSSS) have been installed as backup to the SPING Units. They are used following an accident involving fuel degradation if the SPING monitoring capabilities are lost. Control Terminal CT-1 with System Operator Console CT-1B interrogates each of the SPING and PAVSSS Radiation Monitors for particulate, iodine, and noble gases and informs the operator of changes in operational status, alarm condition, or system parameters within seconds of their occurrence. (ref. 2)

NEI 99-01 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.

Susquehanna Basis Reference(s):

1. NEP Technical Basis 02-005 Rev. #2 Noble Gas Release Rate Limits for EALs
2. FSAR 18.1.30 Accident-Monitoring Instrumentation
4. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.2	**Alert**
Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem child thyroid CDE at or beyond the EMERGENCY PLAN BOUNDARY (Notes 3, 4)	

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

EMERGENCY PLAN BOUNDARY (EPB) - Same as the Exclusion Area Boundary, i.e., that area around SSES within a radius of 1800 feet determined in accordance with 10 CFR 100.11.

Susquehanna Basis:

Dose assessments are performed by computer-based methods (ref. 2).

The EMERGENCY PLAN BOUNDARY (EPB) is used in assessing dose effects to the public rather than the SITE BOUNDARY. The EPB is at or within the SITE BOUNDARY in all compass sectors. The SSES dose projection model (MIDAS) utilizes the EPB when performing dose calculations (ref. 5).

The SITE BOUNDARY is irregularly shaped and therefore would result in dose projections and protective action recommendations that can vary significantly depending on plume direction and affected sector. Using dose projections calculated using the EPB (i.e. Exclusion Area) provides a more consistent approach to Public Protective Action Recommendations since the EPB is more consistently defined in all directions. The EPB is at or within the SITE BOUNDARY in all compass sectors.

The EPA PAGs are expressed in terms of the projected sum of the effective dose equivalent (EDE) from external radiation and the committed effective dose equivalent (CEDE) incurred from inhalation of radioactive materials, or as the committed dose equivalent (CDE) to the thyroid. For the purpose of these IC/EALs, the projected dose quantity Total Effective Dose Equivalent (TEDE), as defined in 10 CFR 20 is used in lieu of "the projected sum of...EDE...and CEDE...", with CEDE considering significant dose from inhaled radionuclides during the early phase of the event. The EPA protective action guidance provides for the use of adult thyroid dose conversion factors. However, the Commonwealth of Pennsylvania requires the use of child thyroid CDE for purposes of comparison of projected thyroid CDE to the PAG for thyroid CDE.

Since dose assessment is based on actual meteorology, whereas the monitor reading RA1.1 is not, the results from these assessments may indicate that the classification is not warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading RA1.1. Classification should not be delayed pending the results of these dose assessments.

NEI 99-01 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.

Susquehanna Basis Reference(s):

1. Susquehanna LLC, Susquehanna Steam Electric Stations Emergency Plan, Section 7.1.1, Off Site Dose Calculations
2. EP-RM-005 SSES MIDAS-NU User Manual
3. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.3 Alert

Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem child thyroid CDE at or beyond the EMERGENCY PLAN BOUNDARY for 60 min. of exposure (Notes 1, 2)

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

EMERGENCY PLAN BOUNDARY (EPB) - Same as the Exclusion Area Boundary, i.e., that area around SSES within a radius of 1800 feet determined in accordance with 10 CFR 100.11.

Susquehanna Basis:

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

For a radiological liquid release, the calculated effluent concentration from a chemistry sample is compared to the emergency action level. Shift Management utilizes emergency response procedures to notify risk counties and to obtain river water samples.

The Susquehanna station incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Technical Requirements Manual (TRM). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of degradation in these features and/or controls.

EAL RA1.3 addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc. This EAL reflects the concern that releases in excess of the referenced offsite dose values represent an uncontrolled situation and hence a potential degradation in the level of safety. Although the calculated dose is very low, it is the degradation in plant control as indicated by the failure to terminate the release that is of primary concern.

The EMERGENCY PLAN BOUNDARY is referenced in EAL RA1.3 because this EAL is based upon liquid release limits from the plant. The release limits are contained in the TRM and are in turn based upon calculation methodology specified in the ODCM. The ODCM utilizes the EMERGENCY PLAN BOUNDARY to establish plant release limits.

NEI 99-01 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. Escalation of the emergency classification level would be via IC RS1.

Susquehanna Basis Reference(s):

1. ODCM-QA-003 Effluent Monitor Setpoints
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.4 Alert

Field survey results indicate **EITHER** of the following at or beyond the EMERGENCY PLAN BOUNDARY:

- Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate child thyroid CDE > 50 mrem for 60 min. of inhalation.

(Notes 1, 2)

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

EMERGENCY PLAN BOUNDARY (EPB) - Same as the Exclusion Area Boundary, i.e., that area around SSES within a radius of 1800 feet determined in accordance with 10 CFR 100.11

Susquehanna Basis:

The EMERGENCY PLAN BOUNDARY (EPB) is used in assessing dose effects to the public rather than the SITE BOUNDARY. The EPB is at or within the SITE BOUNDARY in all compass sectors. The SSES dose projection model (MIDAS) utilizes the EPB when performing dose calculations (ref. 1).

The SITE BOUNDARY is irregularly shaped and therefore would result in dose projections and protective action recommendations that can vary significantly depending on plume direction and affected sector. Using dose projections calculated using the EPB (i.e. Exclusion Area) provides a more consistent approach to Public Protective Action Recommendations since the EPB is more consistently defined in all directions. The EPB is at or within the SITE BOUNDARY in all compass sectors.

The EPA PAGs are expressed in terms of the projected sum of the effective dose equivalent (EDE) from external radiation and the committed effective dose equivalent (CEDE) incurred from inhalation of radioactive materials, or as the committed dose equivalent (CDE) to the thyroid. For the purpose of these IC/EALs, the projected dose quantity Total Effective Dose

Equivalent (TEDE), as defined in 10 CFR 20 is used in lieu of "the projected sum of...EDE...and CEDE...", with CEDE considering significant dose from inhaled radionuclides during the early phase of the event. The EPA protective action guidance provides for the use of adult thyroid dose conversion factors. However, the Commonwealth of Pennsylvania requires the use of child thyroid CDE for purposes of comparison of projected thyroid CDE to the PAG for thyroid CDE.

NEI 99-01 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.

Susquehanna Basis Reference(s):

1. EP-RM-005 SSES MIDAS-NU User Manual
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.1	**Unusual Event**
Gaseous or liquid effluent > Table R-1 column "UE" for ≥ 60 min. (Notes 1, 2, 3)	

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Plant Vent (noble gas)	0C630 0C677	1.9E+09 µCi/min (site total)	1.9E+08 µCi/min (site total)	1.9E+07 µCi/min (site total)	4.0E+06 µCi/min (site total)
Liquid	LRW	RR-06433	----	----	----	2 x hi alarm
Liquid	1(2) RHRSW A/B	RR-D12-1(2)R606	----	----	----	2 x hi alarm
Liquid	1(2) SW/SDHR	RR-D12-1(2)R604	----	----	----	2 x hi alarm

Mode Applicability:

All

Definition(s):

None

Susquehanna Basis:

The EMERGENCY PLAN BOUNDARY is used in RU1.1 because it is based upon release limits from the plant. The release limits are based upon the Plant Technical Requirements Manual and in turn based upon calculation methodology specified in the ODCM. The ODCM utilizes the

EMERGENCY PLAN BOUNDARY to establish plant release limits. EAL RU1.1 is an indication of degradation in the level of safety of the plant.

This EAL represents radioactivity releases, that for whatever reason, cause liquid effluent radiation monitor readings to exceed two times the Technical Requirements Manual limit and releases are not terminated within 60 minutes. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance with the applicable TRM release limit. Indexing the EAL threshold to the ODCM setpoints in this manner ensures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

For a radiological liquid release, the calculated effluent concentration from a chemistry sample is compared to the emergency action level. Shift Management utilizes emergency response procedures to notify risk counties and to obtain river water samples.

<u>Gaseous Releases</u>

The column "UE" gaseous release value in Table R-1 represents two times the appropriate TRM release rate limits (ref. 1-3).

The SPING monitors the radioactive effluent from the Units 1 and 2 Turbine Building and Reactor Building Ventilation Stacks and the Standby Gas Treatment System Exhaust Vent. All five collectively are the Plant Vent on Table R-1. The SPING system is normally aligned to be operated from Brother Control Terminals 0C630 in the Control Room, using 0C677 as a backup in the TSC. Three Post-Accident Vent Stack Sampling Systems (PAVSSS) have been installed as backup to the SPING Units. They are used following an accident involving fuel degradation if the SPING monitoring capabilities are lost. Control Terminal CT-1 with System Operator Console CT-1B interrogates each of the SPING and PAVSSS Radiation Monitors for particulate, iodine, and noble gases and informs the operator of changes in operational status, alarm condition, or system parameters within seconds of their occurrence. (ref. 4)

<u>Liquid Releases</u>

The column "UE" liquid release values in Table R-1 represent two times the appropriate TRM release rate limits associated with the specified release point (ref. 5-9).

- LRW

  A sample pump takes a portion of the LRW effluent line flow and passes it through a scintillation detector (RE-06433). The contents of EDST, the LRW sample tanks, or the laundry drain sample tank can be lined up for release. Release flow is through two isolation valves, HV-06432A1 and A2 to the cooling tower blowdown line.

  The isolation valves will close on:

  - Low Sample Flow (< 0.5 gpm)
  - Low Blowdown Flow (< 5500 gpm)
  - High radiation (calculated for each release based on sample)
  - Rad Monitor downscale (calculated)
  - Rad Monitor inoperable

- RHRSW

  The Residual Heat Removal (RHR) Service Water RMS detects primary coolant leakage into the RHR Service Water during RHR Heat Exchanger operation. Local indication of RHR Service Water Loop 'A' radiation is provided by RITS-11216A and RHR Service Water Loop 'B' by RITS-11216B. Output is also sent to Control Room alarms and Radiation Recorder RR-D12-1R606 on Panel 1C600. The following RHR Service Water RMS high radiation annunciators are located on Control Room Panel 1C601: RHR SW A HI RADIATION (AR-109-F01) is actuated at a setpoint determined by Chemistry.

- RHR SW B HI RADIATION (AR-113-F01) is actuated at a setpoint determined by Chemistry.

- SW/SDHR

  Service Water/Supplemental Decay Heat Removal RMS detects radioactive material in-leakage to the service water system from the spent fuel pool heat exchangers. During unit outages when SDHR is placed in service, the SDHR Detector is connected to the Radiation Monitoring Unit in place of the Service Water Detector.

  The Service Water/Supplemental Decay Heat Removal RMS annunciator SERVICE WATER EFFLUENT HI RADIATION (AR-123-D04) setpoint is variable as determined by Chemistry Group.

NEI 99-01 Basis:

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

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

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

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

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

This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways. EAL Bases Escalation of the emergency classification level would be via IC RA1.

Susquehanna Basis Reference(s):

1. TRM 3.3.4 TRM Post-Accident Monitoring Instrumentation
2. TRM 3.11.2 Gaseous Effluents
3. Susquehanna Calculation EC ENVR 1041 Airborne Effluent Limiting Site Release Rate & Plant Vent Effluent Monitor Setpoints
4. FSAR 18.1.30 Accident-Monitoring Instrumentation
5. OP-179-002 Process Radiation Monitoring System
6. FSAR 9.2 Water Systems
7. TRM 3.11.1.4 Liquid Radwaste Effluent Monitoring Instrumentation
8. TRM 3.11.1.5 Radioactive Liquid Process Monitoring Instrumentation
9. ON-069-001 Abnormal Rad Release Liquid
10. SSES Offsite Dose Calculation Manual
11. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.2	**Unusual Event**
Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x TRM limits for ≥ 60 min. (Notes 1, 2)	

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

None

Susquehanna Basis:

For a radiological liquid release, the calculated effluent concentration from a chemistry sample is compared to the emergency action level. Shift Management utilizes emergency response procedures to notify risk counties and to obtain river water samples.

Limits associated with liquid and gaseous radioactive effluents are contained in the Technical Requirements Manual (TRM). The methodology for calculation of offsite dose or release rates to ensure compliance with the applicable TRM limits is outlined in the ODCM.

EAL RU1.2 addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed two times the applicable TRM limit and are not terminated within 60 minutes. The effluent monitor alarm setpoints are established by the ODCM to warn of a release that is not in compliance with the applicable TRM limit (ref.1). The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. The fundamental basis of this IC is NOT a dose or dose rate, but rather the degradation in the level of safety of the plant implied by the uncontrolled release

EAL RU1.2 includes any release for which a radioactivity discharge permit was not prepared or not applicable, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit. Although the calculated dose is very low, it is the degradation in plant control as indicated by the failure to terminate the release that is of primary concern.

NEI 99-01 Basis:

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

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

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

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

Susquehanna Basis Reference(s):

1. ODCM-QA-003 Effluent Monitor Setpoints
2. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Spent fuel pool level cannot be restored to at least the top of the spent fuel racks for 60 minutes or longer

EAL:

RG2.1	**General Emergency**
Spent fuel pool level **CANNOT** BE RESTORED to at least 0.5 ft. above the top of the spent fuel racks for ≥ 60 min. (Note 1)	

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

All

Definition(s):

None

Susquehanna Basis:

Each fuel storage pool is designed to maintain the water level in the pool above the top of active fuel providing cooling for the fuel rods. Technical Specifications require greater than or equal to 22 ft. of water be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage racks at all times (ref. 1). In the event of loss of fuel pool inventory, Operations will assess multiple indications in accordance with ON-FPC-101(201) and AOP-081-001 (ref. 3, 4).

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal SPF level 22.75 feet above the top of the fuel racks (Level 1 or el. 817'-1"), SFP level 10 ft. above the top of the fuel racks (Level 2 or el. 804'-4") and SFP level at the top of the fuel racks (Level 3 or el. 794'-10" which for SSES is 0.5 feet on instrument above the top of the fuel racks). Each spent fuel pool is equipped with primary and backup guided wave radar probes to measure pool level. The range is continuous from the high pool level elevation (23.25 feet on instrument or el. 817'-7") to the top of the spent fuel racks (0.0 feet on instrument or el. 794'-4") (ref. 2).

Each new SFP LI system provides two alarms to the associated Unit's Control Room Benchboard. The first alarm identifies a low water level condition at 20 ft. (instrument) above the spent fuel rack (elevation 814'-4"). The second alarm identifies a low low water level condition at 10 ft. (instrument) above the spent fuel rack (Level 2 or elevation 804'-4"). These alarms are intended to alert the control room operators of the loss of SFP inventory so actions are taken to *provide make-up as soon as possible* (ref. 2).

NEI 99-01 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.

Susquehanna Basis Reference(s):

1. Technical Specifications 3.7.7 Spent Fuel Storage Pool Water Level
2. PLA-6980 Enclosure 1 Susquehanna Units 1 & 2 Overall Integrated Plan with Regard to Reliable Spent Fuel Pool Instrumentation
3. ON-FPC-101(201) Loss of Fuel Pool Cooling
4. AOP-081-001 Fuel Handling Abnormal Operating Procedure
5. NEI 99-01 AG2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Spent fuel pool level at the top of the fuel racks

EAL:

RS2.1 Site Area Emergency
Lowering of spent fuel pool level to ≤ 0.5 ft. above the top of the spent fuel racks

Mode Applicability:

All

Definition(s):

None

Susquehanna Basis:

Each fuel storage pool is designed to maintain the water level in the pool above the top of active fuel providing cooling for the fuel rods. Technical Specifications require greater than or equal to 22 ft. of water be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage racks at all times (ref. 1). In the event of loss of fuel pool inventory, Operations will assess multiple indications in accordance with ON-FPC-101(201) and AOP-081-001 (ref. 3, 4).

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal SPF level 22.75 feet above the top of the fuel racks (Level 1 or el. 817'-1"), SFP level 10 ft. above the top of the fuel racks (Level 2 or el. 804'-4") and SFP level at the top of the fuel racks (Level 3 or el. 794'-10" which for SSES is 0.5 feet on instrument above the top of the fuel racks). Each spent fuel pool is equipped with primary and backup guided wave radar probes to measure pool level. The range is continuous from the high pool level elevation (23.25 feet on instrument or el. 817'-7") to the top of the spent fuel racks (0.0 feet on instrument or el. 794'-4") (ref. 2).

Each new SFP LI system provides two alarms to the associated Unit's Control Room Benchboard. The first alarm identifies a low water level condition at 20 ft. (instrument) above the spent fuel rack (elevation 814'-4"). The second alarm identifies a low low water level condition at 10 ft. (instrument) above the spent fuel rack (Level 2 or elevation 804'-4"). These alarms are intended to alert the control room operators of the loss of SFP inventory so actions are taken to provide make-up as soon as possible (ref. 2).

NEI 99-01 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.

Susquehanna Basis Reference(s):

1. Technical Specifications 3.7.7 Spent Fuel Storage Pool Water Level
2. PLA-6980 Enclosure 1 Susquehanna Units 1 & 2 Overall Integrated Plan with Regard to Reliable Spent Fuel Pool Instrumentation
3. ON-FPC-101(201) Loss of Fuel Pool Cooling
4. AOP-081-001 Fuel Handling Abnormal Operating Procedure
5. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.1	**Alert**
Uncovery of irradiated fuel in the REFUELING PATHWAY	

Mode Applicability:

All

Definition(s):

REFUELING PATHWAY-. The reactor refueling cavity, spent fuel pool and fuel transfer canal comprise the refueling pathway.

Susquehanna Basis:

This EAL applies to all instances of irradiated fuel handling, including those that are not directly in support of a reactor refueling outage. (e.g. unit-to-unit fuel shuffles, dry fuel storage canister loading, etc.).

Each fuel storage pool is designed to maintain the water level in the pool above the top of active fuel providing cooling for the fuel rods. Technical Specifications require greater than or equal to 22 feet of water be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage racks at all times (ref. 1). In the event of loss of fuel pool inventory, Operations will assess multiple indications in accordance with ON-FPC-101(201) and AOP-081-001 (ref. 3, 4).

NEI 99-01 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. 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 EAL escalates from RU2.1 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 an increase 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.

Escalation of the emergency classification level would be via IC RS1.

Susquehanna Basis Reference(s):

1. Technical Specifications 3.7.7 Spent Fuel Storage Pool Water Level
2. PLA-6980 Enclosure 1 Susquehanna Units 1 & 2 Overall Integrated Plan with Regard to Reliable Spent Fuel Pool Instrumentation
3. ON-FPC-101(201) Loss of Fuel Pool Cooling
4. AOP-081-001 Fuel Handling Abnormal Operating Procedure
5. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity

AND

Any of the following radiation monitor indications:

- Refuel Floor High Exhaust (> 18 mR/hr)
- Refuel Floor Wall Exhaust (> 21 mR/hr)
- Channel 14 Spent Fuel Pool Area Criticality Monitor (> 100 mR/hr)
- Channel 15 Refueling Floor Area (> 80 mR/hr)
- Channel 42 Refueling Floor Area (> 80 mR/hr)
- Channel 47 (U1) / 44 (U2) Spent Fuel Pool Area Criticality Monitor (> 100 mR/hr)
- Channel 49 Refueling Floor High Range Monitor (on scale)

Mode Applicability:

All

Definition(s):

None

Susquehanna Basis:

The Reactor Building ventilation process monitoring system isolates Zone 3 HVAC on high exhaust radiation. Zone 3 exhaust can be monitored at 1C600 (2C600) (ref. 4):

- RR D12 1R605 (2R605), Refuel Floor Wall Exhaust Radiation Monitor
- RR D12 1R607 (2R607), Refuel Floor High Exhaust Radiation Monitor

The listed radiation monitors and specified alarm setpoints/ indications are those associated with a fuel handling accident or damaged spent fuel (ref. 1, 3, 5).

NEI 99-01 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. 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.

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

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).

Escalation of the emergency classification level would be via IC RS1.

Susquehanna Basis Reference(s):

1. OP-179(279)-001 Area Radiation Monitoring System
2. PLA-6980 Enclosure 1 Susquehanna Units 1 & 2 Overall Integrated Plan with Regard to Reliable Spent Fuel Pool Instrumentation
3. AOP-081-001 Fuel Handling Abnormal Operating Procedure
4. EO-000-104 Secondary Containment Control
5. AR-101(201)-001
6. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.3 Alert
Lowering of spent fuel pool level to ≤ 10 ft. above the top of the spent fuel racks

Mode Applicability:

All

Definition(s):

None

Susquehanna Basis:

Each fuel storage pool is designed to maintain the water level in the pool above the top of active fuel providing cooling for the fuel rods. Technical Specifications require greater than or equal to 22 ft. of water be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage racks at all times (ref. 1). In the event of loss of fuel pool inventory, Operations will assess multiple indications in accordance with ON-FPC-101(201) and AOP-081-001 (ref. 3, 4).

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal SPF level 22.75 feet above the top of the fuel racks (Level 1 or el. 817'-1"), SFP level 10 ft. above the top of the fuel racks (Level 2 or el. 804'-4") and SFP level at the top of the fuel racks (Level 3 or el. 794'-10" which for SSES is 0.5 feet on instrument above the top of the fuel racks). Each spent fuel pool is equipped with primary and backup guided wave radar probes to measure pool level. The range is continuous from the high pool level elevation (23.25 feet on instrument or el. 817'-7") to the top of the spent fuel racks (0.0 feet on instrument or el. 794'-4") (ref. 2).

Each new SFP LI system provides two alarms to the associated Unit's Control Room Benchboard. The first alarm identifies a low water level condition at 20 ft. (instrument) above the spent fuel rack (elevation 814'-4"). The second alarm identifies a low low water level condition at 10 ft. (instrument) above the spent fuel rack (Level 2 or elevation 804'-4"). These alarms are intended to alert the control room operators of the loss of SFP inventory so actions are taken to provide make-up as soon as possible (ref. 2).

NEI 99-01 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. 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.

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.

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 classification level would be via IC RS1.

Susquehanna Basis Reference(s):

1. Technical Specifications 3.7.7 Spent Fuel Storage Pool Water Level
2. PLA-6980 Enclosure 1 Susquehanna Units 1 & 2 Overall Integrated Plan with Regard to Reliable Spent Fuel Pool Instrumentation
3. ON-FPC-101(201) Loss of Fuel Pool Cooling
4. AOP-081-001 Fuel Handling Abnormal Operating Procedure
5. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Unplanned loss of water level above irradiated fuel

EAL:

RU2.1 Unusual Event

UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **any** of the following on **EITHER** unit:

- Fuel Pool Water Low Level alarm
- Skimmer Surge Tank Low Level alarm
- Visual observation of a water level drop below a fuel pool skimmer surge tank inlet
- Observation of water draining down the outside wall of primary containment

AND

UNPLANNED rise in area radiation levels as indicated by **any** of the following radiation monitors:

- Channel 14 Spent Fuel Pool Area Criticality Monitor
- Channel 15 Refueling Floor Area
- Channel 42 Refueling Floor Area

Mode Applicability:

All

Definition(s):

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.

REFUELING PATHWAY-. The reactor refueling cavity, spent fuel pool and fuel transfer canal comprise the refueling pathway.

Susquehanna Basis:

Each fuel storage pool is designed to maintain the water level in the pool above the top of active fuel providing cooling for the fuel rods. Technical Specifications require greater than or equal to 22 feet of water be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage racks at all times (ref. 1). In the event of loss of fuel pool inventory, Operations will assess multiple indications in accordance with ON-FPC-101(201) and AOP-081-001 (ref. 4, 6).

When the spent fuel pool and reactor cavity are connected, there could exist the possibility of uncovering irradiated fuel. Therefore, this EAL is applicable for conditions in which irradiated fuel is being transferred to and from the RPV and spent fuel pool.

Control room alarms associated with elevated refuel floor area radiation levels include (ref. 4, 5):

- REFUELING FLOOR AREA HI RADIATION (AR-101-D05)
- SPENT FUEL POOL AREA HI RADIATION (AR-101-E05)

The listed ARMs are the normal range monitors that detect increasing area radiation due to a lack of shielding in the REFUELING PATHWAY (ref. 6).

NEI 99-01 Basis:

This IC addresses a decrease 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 decrease 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 (if available). A significant drop in the water level may also cause an increase in the radiation levels of adjacent areas that can be detected by monitors in those locations.

The effects of planned evolutions should be considered. For example, a refueling bridge area radiation monitor reading may increase 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.

Susquehanna Basis Reference(s):

1. Technical Specifications 3.7.7 Spent Fuel Storage Pool Water Level
2. OP-179(279)-001 Area Radiation Monitoring System
3. AOP-081-001 Fuel Handling Abnormal Operating Procedure
4. EO-000-104 Secondary Containment Control
5. SSES-FSAR Table 12.3-7 Area Radiation Monitoring System
6. ON-FPC-101(201) Loss of Fuel Pool Cooling
7. NEI 99-01 AU2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

RA3.1 Alert

Dose rates > 15 mR/hr in **any** of the following areas:

- Main Control Room
- Radwaste Control Room
- Both the Central Alarm Station (CAS) and Secondary Alarm Station (SAS)

Mode Applicability:

All

Definition(s):

None

Susquehanna Basis:

Central Alarm Station (CAS) and Secondary Alarm Station (SAS) are included in this EAL because of their importance in permitting access to areas required to assure safe plant operations (ref. 1). Both are included in this EAL because either security station can effectively permit access to areas required to assure safe plant operations. It is not the intent of this EAL that there be continuous radiation monitoring in the CAS or SAS. However, if a radiological release is in progress and there are indications that the release may affect the CAS and SAS and dose rates in both areas are determined by manual radiological survey to be greater than 15 mR/hr, an Alert shall be declared.

NEI 99-01 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/Recovery Manager should consider the cause of the increased radiation levels and determine if another IC may be applicable.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

Susquehanna Basis Reference(s):

1. AR-1650683
2. NEI 99-01 AA3

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

RA3.2 Alert

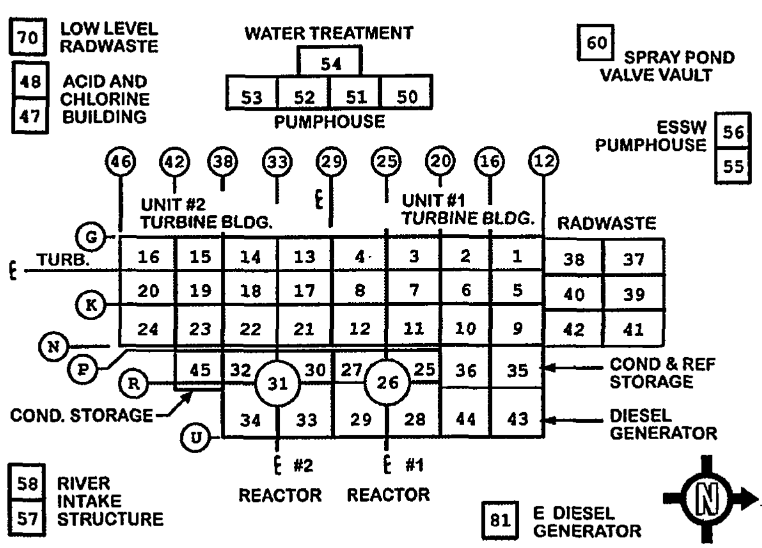
An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 rooms or areas (Note 5)

Note 5: 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.

Table R-2 Safe Operation & Shutdown Areas			
Elevation	**Unit 1 Area(s) ****	**Unit 2 Area(s) ****	**Mode(s)**
670'	RB 27	RB 32	3,4,5
683'	RB 27, 28, 29	RB 32, 33, 34	3,4,5
703'	RB 28, 29	RB 33, 34	3,4,5
719'	RB 25, 29	RB 30, 34	3,4,5
749'	RB 25, 29	RB 32, 33	3,4,5

**** See Chart 1 for location of plant areas**

Chart 1- Plant Area Key Plan



Mode Applicability:

3 - Hot Shutdown

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective measures such as temporary shielding, SCBAs or beyond Emergency Plan RWP dose extensions that are not routinely employed to access the room/area).

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.

Susquehanna Basis:

If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

The list of plant areas in Table R-2 specify those areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area. See Chart 1 for the specific locations of areas listed in Table R-2. See Attachment 3 for more details of how the Table R-2 was developed (ref. 1).

NEI 99-01 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/Recovery Manager should consider the cause of the increased radiation levels and determine if another IC may be applicable.

For RA3.2, 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 increased 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 increase occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.

- The increased 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.
- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

Susquehanna Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases
2. NEI 99-01 AA3

Category C – Cold Shutdown / Refueling System Malfunction

EAL Group: Cold Conditions (RCS temperature ≤ 200°F); EALs in this category are applicable only in one or more cold operating modes.

Category C EALs are directly associated with cold shutdown or refueling system safety functions. Given the variability of plant configurations (e.g., systems out-of-service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The cold shutdown and refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, CONTAINMENT CLOSURE, and fuel clad integrity for the applicable operating modes (4 - Cold Shutdown, 5 - Refueling, D – Defueled).

The events of this category pertain to the following subcategories:

1. RPV Level

Reactor Pressure Vessel water level is directly related to the status of adequate core cooling and, therefore, fuel clad integrity.

2. Loss of Essential AC Power

Loss of essential plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite power sources for 4.16 kV ESS buses.

3. Loss of Vital DC Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of power to or degraded voltage on the 125 VDC vital buses.

4. RCS Temperature

Uncontrolled or inadvertent temperature or pressure increases are indicative of a potential loss of safety functions.

5. Loss of Communications

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

6. Hazardous Event Affecting Safety Systems

Certain hazardous natural and technological events may result in visible damage to or degraded performance of safety systems warranting classification.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Loss of RPV inventory affecting fuel clad integrity with Containment challenged

EAL:

CG1.1	**General Emergency**
RPV level < -161 in. (TAF) for ≥ 30 min. (Note 1)	
AND	
Any Containment Challenge indication, Table C-2	

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is **not** required.

Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6)
• PC hydrogen concentration > 6%
• UNPLANNED rise in PC pressure
• Exceeding one or more Secondary Containment Control Max Safe Radiation Levels (EO-000-104 Table 9) that can be read in the Control Room (Table C-6)

Table C-6

Max Safe Reactor Building Radiation Limits

RB Area Elevation (ft)	ARM Number	ARM Channel Description	Max Safe Rad Limit (R/HR)
818	49	Refuel Floor Area	10
749	52 54	RWCU Recirc PP Access Fuel Pool PP Area	10
719	50 51	CRD North CRD South	10
670	53	Remote Shutdown Room	10
645	48 57 55 56	HPCI PP & Turbine Room RCIC PP & Turbine Room RHR A C PP Room RHR B D PP Room	10

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to Susquehanna, CONTAINMENT CLOSURE is established per NDAP-QA-0309 (ref. 4) for Primary Containment OR is established per NDAP-QA-0321 (ref. 5) for Secondary Containment.

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.

Susquehanna Basis:

When RPV level drops below -161 in., core uncovery starts to occur (ref. 1).

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

- CONTAINMENT CLOSURE is not established.

- In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gases in the Primary Containment. However, Primary Containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. An explosive mixture can be formed when hydrogen gas concentration in the Primary Containment atmosphere is greater than 6% by volume in the presence of oxygen (>5%) (ref. 2). In Cold Shutdown and Refueling modes it is assumed that the Primary Containment is de-inerted.
- Any unplanned increase in PC pressure in the Cold Shutdown or Refueling mode indicates a potential loss of CONTAINMENT CLOSURE capability. Unplanned Primary Containment pressure increases indicates CONTAINMENT CLOSURE cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.
- Secondary Containment radiation monitors that can be read in the Control Room should provide indication of increased release that may be indicative of a challenge to CONTAINMENT CLOSURE. The Max Safe radiation levels are indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in EO-000-104, Secondary Containment Control, (ref. 3).

NEI 99-01 Basis:

This IC addresses the inability to restore and maintain reactor vessel 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 reactor vessel level. If RPV 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 the other listed indications to assess whether or not containment is challenged.

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.

This EAL addresses 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.*

Susquehanna Basis Reference(s):

1. EO-000-101 RPV Control
2. EP-DS-001 Containment Combustible Gas Control
3. EO-000-104 Secondary Containment Control
4. NDAP-QA-0309 Primary Containment Access and Control
5. NDAP-QA-0321 Secondary Containment Integrity Control
6. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Loss of RPV inventory affecting fuel clad integrity with Containment challenged

EAL:

CG1.2 General Emergency

RPV level **cannot** be monitored for ≥ 30 min. (Note 1)

AND

UNPLANNED increase in **any** Table C-1 sump or tank level due to a loss of RPV inventory of sufficient magnitude to indicate core uncovery

AND

Any Containment Challenge indication, Table C-2

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is **not** required.

Table C-1 Sumps & Tanks
• Drywell equipment drain tank • Drywell sumps • Reactor Building sump • LRW collection tanks • Main condenser hotwell • Suppression pool • Visual observation

Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • PC hydrogen concentration > 6% • UNPLANNED rise in PC pressure • Exceeding one or more Secondary Containment Control Max Safe Radiation Levels (EO-000-104 Table 9) that can be read in the Control Room (Table C-6)

Table C-6

Max Safe Reactor Building Radiation Limits

RB Area Elevation (ft)	ARM Number	ARM Channel Description	Max Safe Rad Limit (R/HR)
818	49	Refuel Floor Area	10
749	52 54	RWCU Recirc PP Access Fuel Pool PP Area	10
719	50 51	CRD North CRD South	10
670	53	Remote Shutdown Room	10
645	48 57 55 56	HPCI PP & Turbine Room RCIC PP & Turbine Room RHR A C PP Room RHR B D PP Room	10

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

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.

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to Susquehanna, CONTAINMENT CLOSURE is established per NDAP-QA-0309 (ref. 7) for Primary Containment OR is established per NDAP-QA-0321 (ref. 8) for Secondary Containment.

Susquehanna Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. RPV level in the Refueling mode is normally monitored using the Shutdown Range or temporary RPV shutdown level transmitter (ref. 1).

If RPV level monitoring capability is unavailable, the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Rise in drywell equipment drain tank level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 2). A Reactor Building sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage (ref. 3, 4). If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

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

- CONTAINMENT CLOSURE is not established.
- In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gases in the Primary Containment. However, Primary Containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. An explosive mixture can be formed when hydrogen gas concentration in the Primary Containment atmosphere is greater than 6% by volume in the presence of oxygen (>5%) (ref. 5). In Cold Shutdown and Refueling modes it is assumed that the Primary Containment is de-inerted.
- Any unplanned increase in PC pressure in the Cold Shutdown or Refueling mode indicates a potential loss of CONTAINMENT CLOSURE capability. Unplanned Primary Containment pressure increases indicates CONTAINMENT CLOSURE cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.
- Secondary Containment radiation monitors should provide indication of increased release that may be indicative of a challenge to CONTAINMENT CLOSURE. The Max Safe radiation levels are indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in EO-000-104, Secondary Containment Control, (ref. 6).

NEI 99-01 Basis:

This IC addresses the inability to restore and maintain reactor vessel 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 reactor vessel level. If RPV 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 the other listed indications to assess whether or not containment is challenged.

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 RPV 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.

This EAL addresses 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.*

Susquehanna Basis Reference(s):

1. IC 180 005 Installation and Removal of Temporary Unit 1 RPV Shutdown Level Transmitter
2. ON-100(200)-005 Excess Drywell Leakage Identification
3. OP-149(249)-002 RHR Operation in Shutdown Cooling Mode
4. ON-149(249)-001 Loss of RHR Shutdown Cooling Mode
5. EP-DS-001 Containment Combustible Gas Control
6. EO-000-104 Secondary Containment Control
7. NDAP-QA-0309 Primary Containment Access and Control
8. NDAP-QA-0321 Secondary Containment Integrity Control
9. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.1	**Site Area Emergency**
CONTAINMENT CLOSURE **not** established **AND** RPV level < -129 in. (Level 1)	

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to Susquehanna, CONTAINMENT CLOSURE is established per NDAP-QA-0309 (ref. 3) for Primary Containment OR is established per NDAP-QA-0321 (ref. 4) for Secondary Containment.

Susquehanna Basis:

RPV level is normally monitored using the instruments in Figure C-1 (ref. 1, 2).

When RPV level decreases to -129 in., RPV water level is below the low-low-low ECCS actuation setpoint (Level 1) (ref. 1).

The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV water level decrease and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier.

NEI 99-01 Basis:

This IC addresses a significant and prolonged loss of RCS level 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 reactor vessel level. If RPV level cannot be restored, fuel damage is probable.

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions.

The difference in the specified RPV levels of CS1.1 and CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

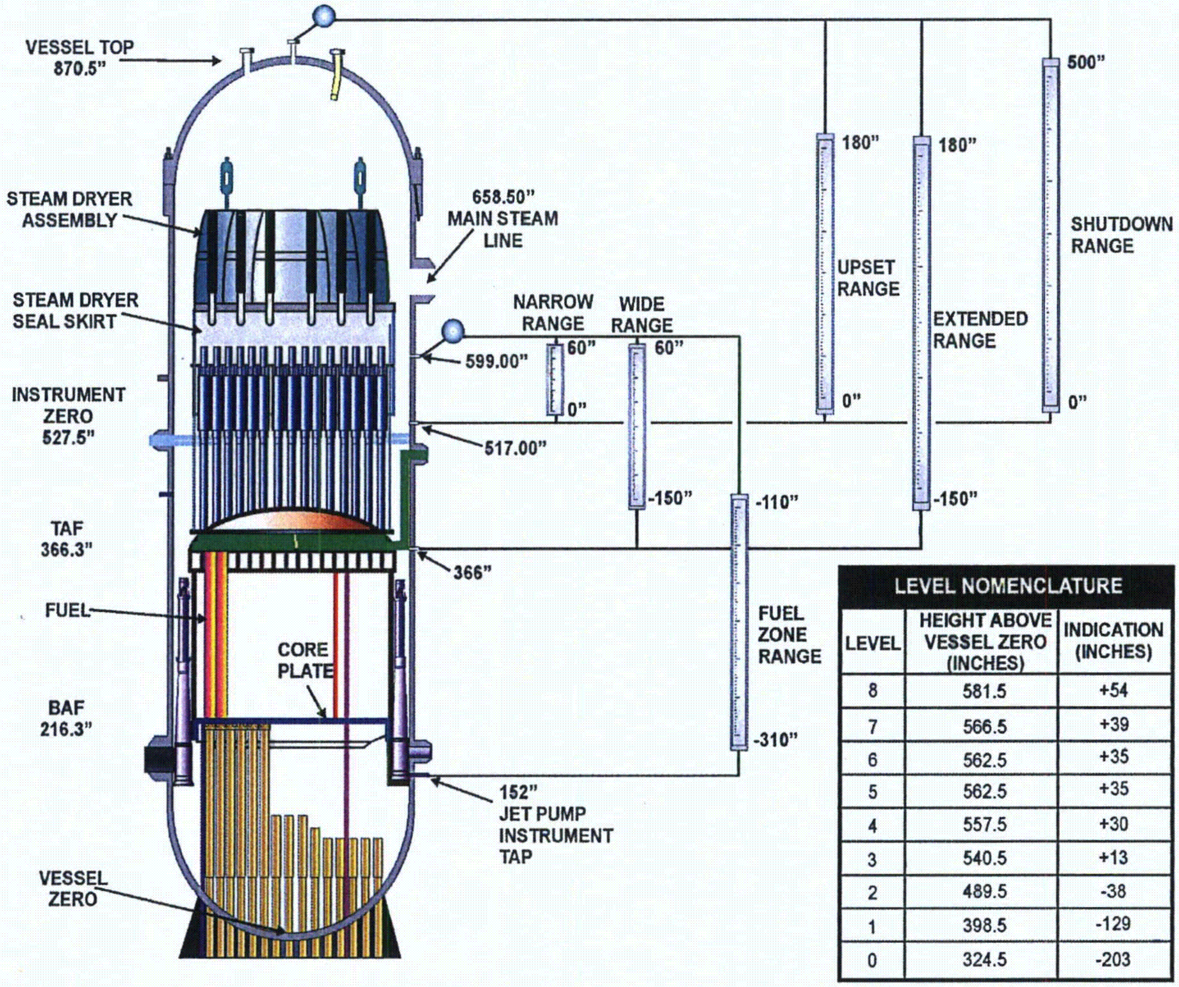
This EAL addresses concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal; SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues; NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States; and NUMARC 91-06, Guidelines for Industry Actions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

Escalation of the emergency classification level would be via IC CG1 or RG1.

Susquehanna Basis Reference(s):

1. M-142 P&ID Nuclear Boiler Vessel Instrumentation, Sheets 1, 2
2. ON-145(245)-004 RPV Water Level Anomaly
3. NDAP-QA-0309 Primary Containment Access and Control
4. NDAP-QA-0321 Secondary Containment Integrity Control
5. NEI 99-01 CS1

Figure C-1 RPV Levels (ref. 1, 2)



LEVEL NOMENCLATURE		
LEVEL	HEIGHT ABOVE VESSEL ZERO (INCHES)	INDICATION (INCHES)
8	581.5	+54
7	566.5	+39
6	562.5	+35
5	562.5	+35
4	557.5	+30
3	540.5	+13
2	489.5	-38
1	398.5	-129
0	324.5	-203

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.2	**Site Area Emergency**
CONTAINMENT CLOSURE established **AND** RPV level < -161 in. (TAF)	

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to Susquehanna, CONTAINMENT CLOSURE is established per NDAP-QA-0309 (ref. 2) for Primary Containment OR is established per NDAP-QA-0321 (ref. 3) for Secondary Containment.

Susquehanna Basis:

When RPV level drops to the top of active fuel (TAF) (an indicated RPV level of -161 in.), core uncovery starts to occur (ref. 1).

NEI 99-01 Basis:

This IC addresses a significant and prolonged loss of RPV level 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 reactor vessel level. If RPV level cannot be restored, fuel damage is probable.

Outage/shutdown contingency plans typically 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/reactor vessel levels of CS1.1 and CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

This EAL addresses 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 or RG1.

Susquehanna Basis Reference(s):

1. EO-000-102 RPV Control
2. NDAP-QA-0309 Primary Containment Access and Control
3. NDAP-QA-0321 Secondary Containment Integrity Control
4. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.3 Site Area Emergency

RPV level cannot be monitored for ≥ 30 min. (Note 1)
AND
UNPLANNED increase in **any** Table C-1 sump or tank level due to a loss of RPV inventory of sufficient magnitude to indicate core uncovery

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-1 Sumps & Tanks
• Drywell equipment drain tank • Drywell sumps • Reactor Building sump • LRW collection tanks • Main condenser hotwell • Suppression pool • Visual observation

Mode Applicability:

4 – Cold Shutdown, 5 – Refueling

Definition(s):

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.

Susquehanna Basis:
In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. RPV level in the Refueling mode is normally monitored using the Shutdown Range or temporary RPV shutdown level transmitter (ref. 1).

If RPV level monitoring capability is unavailable, the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage.

Rise in drywell equipment drain tank level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 2). A Reactor Building sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage (ref. 3, 4). If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

NEI 99-01 Basis:

This IC addresses a significant and prolonged loss of RPV 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 reactor vessel level. If RPV level cannot be restored, fuel damage is probable.

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 RPV 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.

This EAL addresses 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 or RG1

Susquehanna Basis Reference(s):

1. IC 180 005 Installation and Removal of Temporary Unit 1 RPV Shutdown Level Transmitter
2. ON-100(200)-005 Excess Drywell Leakage Identification
3. OP-149(249)-002 RHR Operation in Shutdown Cooling Mode
4. ON-149(249)-001 Loss of RHR Shutdown Cooling Mode
5. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Loss of RPV inventory

EAL:

CA1.1	**Alert**
Loss of RPV inventory as indicated by RPV level < -38 in. (Level 2)	

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

None

Susquehanna Basis:

The threshold RPV level of -38 in. is the low-low ECCS actuation setpoint (ref. 1). RPV level is normally monitored using the instruments in Figure C-1 (ref. 1, 2).

When reactor vessel water level drops to -38 in. high pressure steam-driven injection sources HPCI (ECCS) and RCIC receive an initiation signal (ref. 1, 2). Although these systems cannot restore RCS inventory in the cold condition, the Level 2 actuation setpoint is operationally significant and is indicative of a loss of RCS inventory significantly below the low level scram setpoint specified in CU1.1.

NEI 99-01 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.

For this EAL, a lowering of water level below - 38 in. indicates that operator actions have not been successful in restoring and maintaining RPV water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

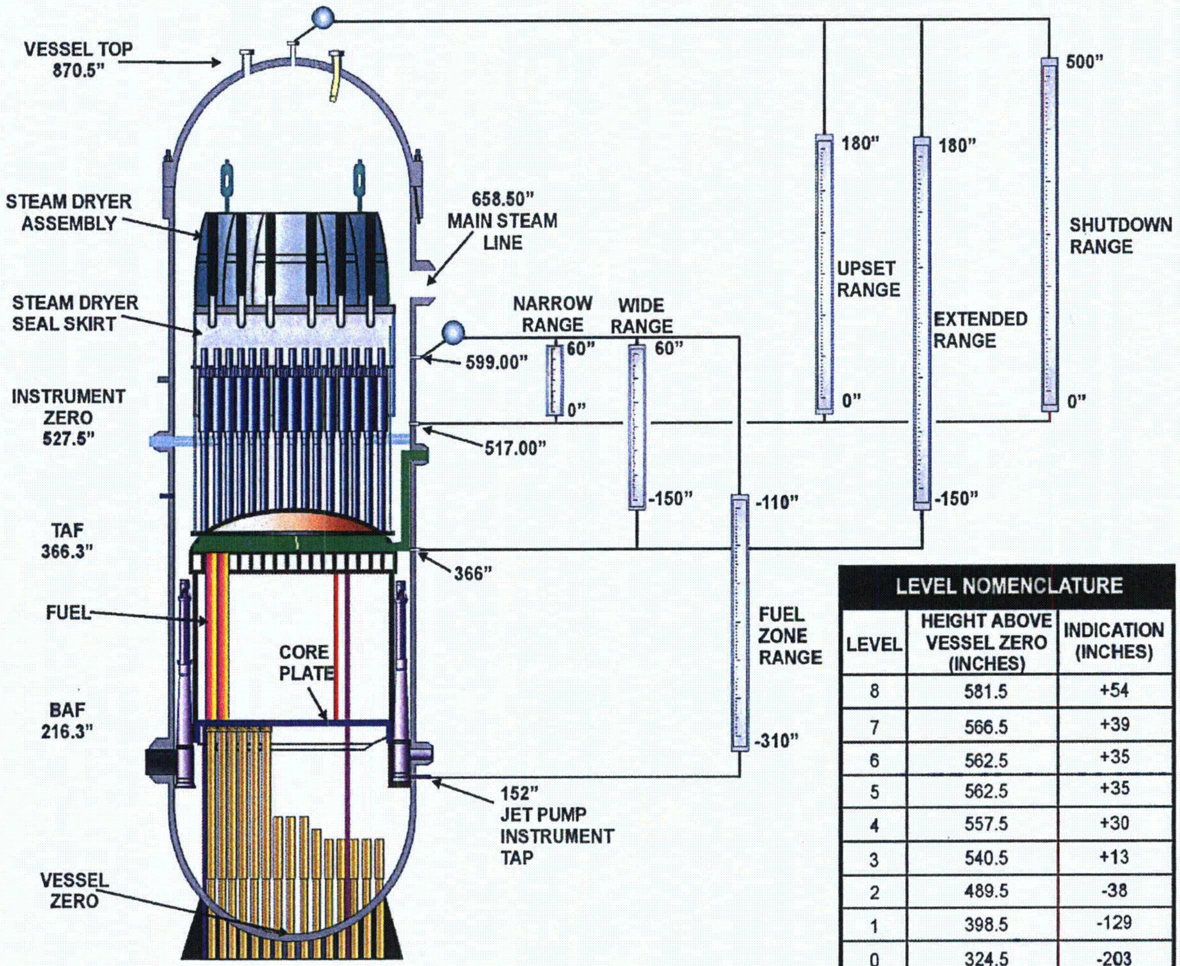
Although related, this EAL is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Residual Heat Removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA4.

If RPV water level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

Susquehanna Basis Reference(s):

1. M-142 P&ID Nuclear Boiler Vessel Instrumentation, Sheets 1, 2
2. ON-145(245)-004 RPV Water Level Anomaly
3. NEI 99-01 CA1

Figure C-1 RPV Levels (ref. 1, 2)



LEVEL NOMENCLATURE		
LEVEL	HEIGHT ABOVE VESSEL ZERO (INCHES)	INDICATION (INCHES)
8	581.5	+54
7	566.5	+39
6	562.5	+35
5	562.5	+35
4	557.5	+30
3	540.5	+13
2	489.5	-38
1	398.5	-129
0	324.5	-203

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Loss of RPV inventory

EAL:

CA1.2 **Alert**
RPV level cannot be monitored for ≥ 15 min. (Note 1)
<u>AND</u>
UNPLANNED increase in **any** Table C-1 sump or tank levels due to a loss of RPV inventory

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-1 Sumps & Tanks
• Drywell equipment drain tank
• Drywell sumps
• Reactor Building sump
• LRW collection tanks
• Main condenser hotwell
• Suppression pool
• Visual observation

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

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.

Susquehanna Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. RPV level in the Refueling mode is normally monitored using the Shutdown Range or temporary RPV shutdown level transmitter (ref. 1).

In this EAL, all water level indication is unavailable for greater than 15 minutes and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Rise in drywell equipment drain tank level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 2). A Reactor Building sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With

RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage (ref. 3, 4). If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

NEI 99-01 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.

For this EAL, the inability to monitor RPV 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 RPV.

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 RPV inventory level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

Susquehanna Basis Reference(s):

1. IC 180 005 Installation and Removal of Temporary Unit 1 RPV Shutdown Level Transmitter
2. ON-100(200)-005 Excess Drywell Leakage Identification
3. OP-149(249)-002 RHR Operation in Shutdown Cooling Mode
4. ON-149(249)-001 Loss of RHR Shutdown Cooling Mode
5. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CU1.1	**Unusual Event**
UNPLANNED *loss of reactor coolant results in* RPV *level less than a required lower limit for* ≥ 15 min. (Note 1)	

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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.

Susquehanna Basis:

Figure C-1 illustrates the elevations of the RPV level instrument ranges (ref. 1).

With the plant in Cold Shutdown, RPV water level is normally maintained above the RPV low level scram setpoint of +13 in. (ref. 1, 2). However, if RPV level is being controlled below the RPV low level scram setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

With the plant in Refueling mode, RPV water level is normally maintained at or above the reactor vessel flange (Technical Specifications 3.9.6 requires at least 22 ft of water above the top of the reactor vessel flange in the refueling cavity during refueling operations). The RPV flange is at an indicated level of 217.5 in. as indicated on the Shutdown Range RPV water level instrument (ref. 3).

EAL RU2.1 may also be applicable based on increasing radiation levels due to loss of inventory in the REFULING PATHWAY.

NEI 99-01 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 RPV 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 decrease 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.

This EAL recognizes that the minimum required RPV 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 typically specified in the applicable operating procedure but may be specified in another controlling document.

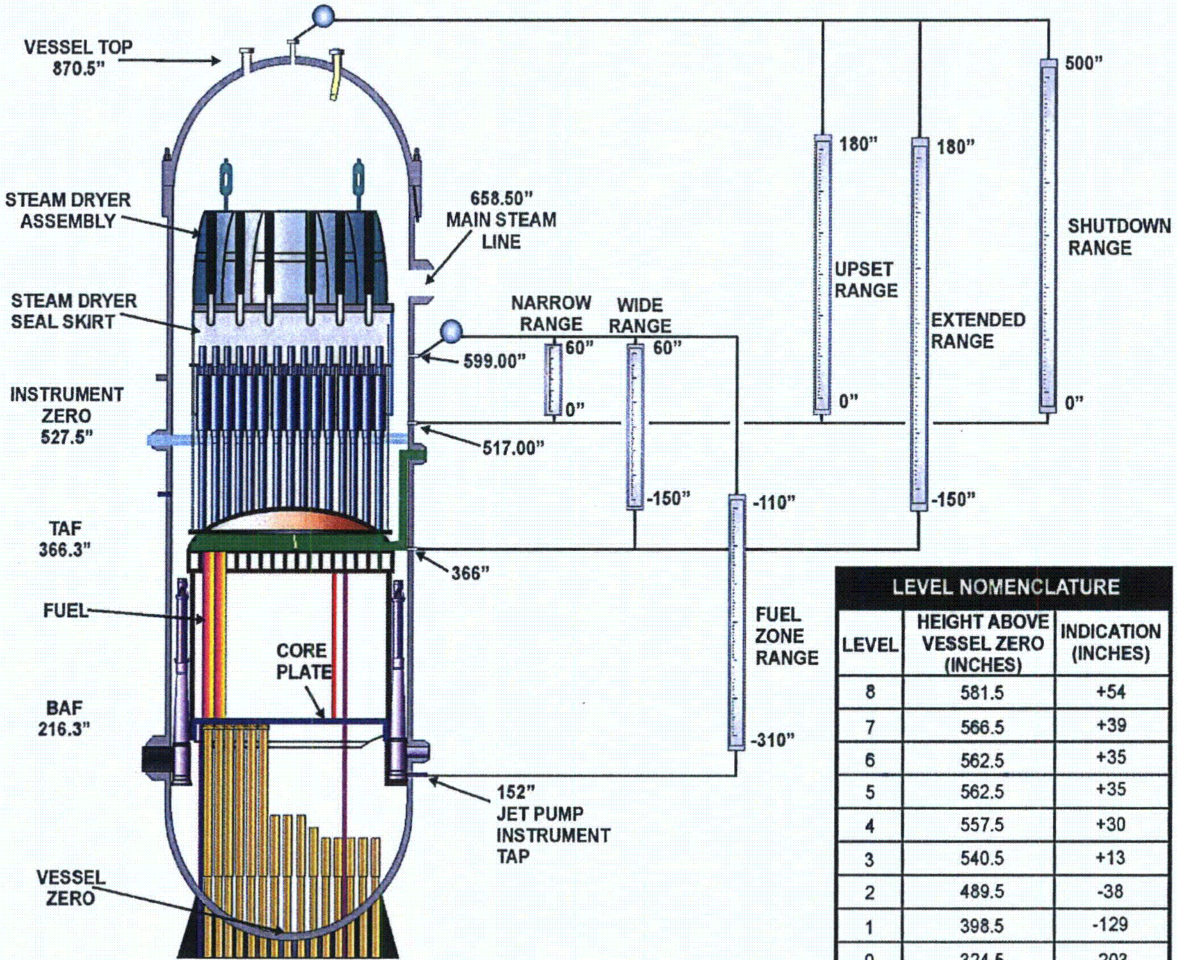
The 15-minute threshold duration allows sufficient time for prompt operator actions to restore and maintain the expected water level. This criterion excludes transient conditions causing a brief lowering of water level.

Continued loss of RCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA3.

Susquehanna Basis Reference(s):

1. M-142 P&ID Nuclear Boiler Vessel Instrumentation, Sheets 1, 2
2. ON-145(245)-004 RPV Water Level Anomaly
3. GO-100(200)-006 Cold Shutdown, Refueling and Defueled
5. NEI 99-01 CU1

Figure C-1 RPV Levels (ref. 1, 2)



LEVEL NOMENCLATURE		
LEVEL	HEIGHT ABOVE VESSEL ZERO (INCHES)	INDICATION (INCHES)
8	581.5	+54
7	566.5	+39
6	562.5	+35
5	562.5	+35
4	557.5	+30
3	540.5	+13
2	489.5	-38
1	398.5	-129
0	324.5	-203

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CU1.2 Unusual Event

RPV water level cannot be monitored

AND

UNPLANNED increase in **any** Table C-1 sump or tank levels due to a loss of RPV inventory

Table C-1 Sumps & Tanks
• Drywell equipment drain tank • Drywell sumps • Reactor Building sump • LRW collection tanks • Main condenser hotwell • Suppression pool • Visual observation

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

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.

Susquehanna Basis:

Note: *The 15 minute criteria in the IC only applies to* EAL CU1.1. *If the conditions specified in* this EAL last longer than 15 minutes, CA1.2 applies.

In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. RPV level in the Refueling mode is normally monitored using the Shutdown Range or temporary RPV shutdown level transmitter (ref. 1).

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Rise in drywell equipment drain tank level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 2). A Reactor Building sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage (ref. 3, 4). If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

NEI 99-01 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 RPV 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 decrease 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.

This EAL addresses a condition where all means to determine RPV 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 RPV.

Continued loss of RCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA4.

Susquehanna Basis Reference(s):

1. IC 180 005 Installation and Removal of Temporary Unit 1 RPV Shutdown Level Transmitter
2. ON-100(200)-005 Excess Drywell Leakage Identification
3. OP-149(249)-002 RHR Operation in Shutdown Cooling Mode
4. ON-149(249)-001 Loss of RHR Shutdown Cooling Mode
5. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CA2.1	**Alert**
Loss of **ALL** offsite and **ALL** onsite AC power capability to **ALL** 4.16 kV ESS buses on **EITHER** unit for ≥ 15 min. (Note 1)	

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

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

Susquehanna Basis:

The Class 1E 4.16 kV system supplies all the Engineered Safety Feature (ESF) loads and other loads that are needed for a safe and orderly plant shutdown, and for keeping the plant in a safe shutdown condition. See Figure C-2 (ref. 1, 2) The eight Class 1E 4.16 kV ESS Buses 1(2)A through 1(2)D receive power from either the four ESS 13.8/4.16 kV transformers or the diesel generators (A, B, C, D and additional diesel generator E). Buses 1A-1D supply Unit 1 and common loads and Buses 2A-2D supply Unit 2 loads. This configuration prevents a loss of all ESS Buses for one unit in the event one of the ESS Transformers is lost.

During normal plant operation, ESS Transformer 101 supplies preferred power to ESS Bus 1A and 2A and is an alternate power supply to ESS bus 1D and 2D. ESS Transformer 111 supplies preferred power to ESS Bus 1C and 2C, and is an alternate power supply to ESS bus 1B and 2B. ESS Transformer 201 supplies preferred power to ESS Bus 1D and 2D, and is an alternate power supply to ESS bus 1A and 2A. ESS Transformer 211 supplies preferred power to ESS Bus 1B and 2B, and is an alternate power supply to ESS bus 1C and 2C.

On a loss of a preferred power source, the bus rapidly transfers to the alternate power source to maintain component power. If both the preferred and alternate power sources are lost, the associated standby diesel generator connects to the ESS bus. (ref. 2-6)

This cold condition EAL is equivalent to the hot condition loss of all offsite AC power EAL SS1.1.

NEI 99-01 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 increased 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.

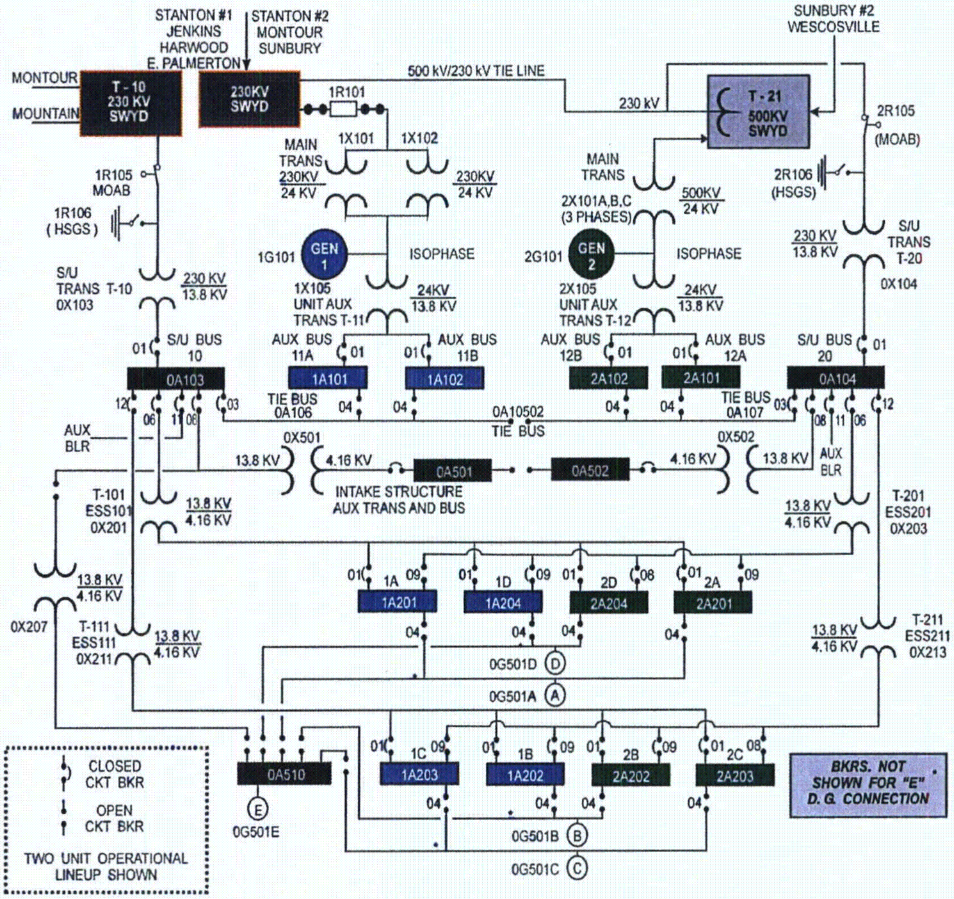
Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via IC CS1 or RS1.

Susquehanna Basis Reference(s):

1. FSAR Section 8.2 Offsite Power System
2. FSAR Section 8.3 Onsite Power System
3. Technical Specifications 3.8.2 AC Sources – Shutdown
4. Technical Specifications 3.8.8 Distribution System – Shutdown
5. ON-104 (204)-001 Units 1(2) Response to Loss of All Offsite Power
6. EO-100 (200)-030 UNIT 1(2) Response to Station Blackout
7. NEI 99-01 CA2

Figure C-2 ESS 13.8/4.16 kV Transformers and Distribution (ref. 1)



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** but one AC power source to essential buses for 15 minutes or longer

EAL:

CU2.1 Unusual Event

AC power capability to **ALL** 4.16 kV ESS buses on **EITHER** unit reduced to a single power source for ≥ 15 min. (Note 1)

AND

Any additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

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

Definition(s):

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 (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Susquehanna Basis:

The Class 1E 4.16 kV system supplies all the Engineered Safety Feature (ESF) loads and other loads that are needed for a safe and orderly plant shutdown, and for keeping the plant in a safe shutdown condition. See Figure C-2 (ref. 1, 2) The eight Class 1E 4.16 kV ESS Buses 1(2)A through 1(2)D receive power from either the four ESS 13.8/4.16 kV transformers or the diesel generators (A, B, C, D and additional diesel generator E). Buses 1A-1D supply Unit 1 and common loads and Buses 2A-2D supply Unit 2 loads. This configuration prevents a loss of all ESS Buses for one unit in the event one of the ESS Transformers is lost.

During normal plant operation, ESS Transformer 101 supplies preferred power to ESS Bus 1A and 2A and is an alternate power supply to ESS bus 1D and 2D. ESS Transformer 111 supplies preferred power to ESS Bus 1C and 2C, and is an alternate power supply to ESS bus 1B and 2B. ESS Transformer 201 supplies preferred power to ESS Bus 1D and 2D, and is an alternate power supply to ESS bus 1A and 2A. ESS Transformer 211 supplies preferred power to ESS Bus 1B and 2B, and is an alternate power supply to ESS bus 1C and 2C.

On a loss of a preferred power source, the bus rapidly transfers to the alternate power source to maintain component power. If both the preferred and alternate power sources are lost, the associated standby diesel generator connects to the ESS bus. (ref. 2-6)

This *cold condition* EAL is *equivalent to the hot condition* EAL SA1.1.

NEI 99-01 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 increased 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 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 one division of emergency power sources (e.g., onsite diesel generators).
- A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from the unit main generator.
- A loss of emergency power sources (e.g., onsite diesel generators) with a single division of emergency buses being back-fed from an offsite power source.

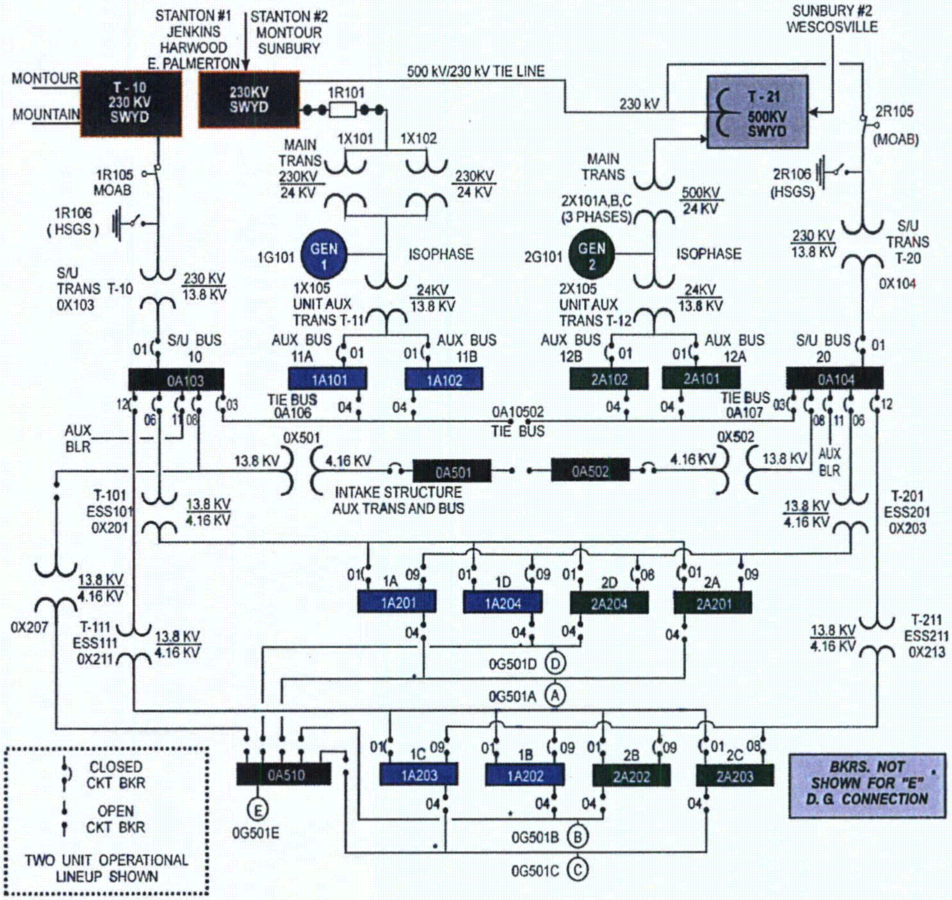
Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

The subsequent loss of the remaining single power source would escalate the event to an Alert in accordance with IC CA2.

Susquehanna Basis Reference(s):

1. FSAR Section 8.2 Offsite Power System
2. FSAR Section 8.3 Onsite Power System
3. Technical Specifications 3.8.2 AC Sources – Shutdown
4. Technical Specifications 3.8.8 Distribution System – Shutdown
5. ON-104 (204)-001 Units 1(2) Response to Loss of All Offsite Power
6. EO-100 (200)-030 UNIT 1(2) Response to Station Blackout
7. NEI 99-01 CU2

Figure C-2 ESS 13.8/4.16 kV Transformers and Distribution (ref. 1)



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – Loss of Vital DC Power

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

EAL:

CU3.1	**Unusual Event**
< 105 VDC bus voltage indications on Technical Specification **required** 125 VDC buses for ≥ 15 min. (Note 1)	

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

None

Susquehanna Basis:

The Class 1E Battery Banks are 1(2)D610 (Channel A), 1(2)D620 (Channel B), 1(2)D630 (Channel C), and 1(2)D640 (Channel D). Each bank consists of 60 cells connected in series. Each cell produces a nominal voltage of 2.06 VDC resulting in a total battery bank terminal voltage of 123.6 VDC. All battery banks are designed to supply power to its load center for four hours in the event of a loss of power from its battery charger (ref. 1-3).

105 VDC is the minimum design voltage limit (ref. 4).

Indicated voltage for the vital 125 VDC main distribution buses is local only. Local voltage indication is available for each bus based on dispatching a field operator in accordance with Control Room alarm response procedure AR-1(2)06-001 (A12,B12,C12,D12). Field observation of indicated voltage constitutes the point in time when availability of indications to plant operators that an emergency action level has been, or may be, exceeded.

This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS2.1.

NEI 99-01 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 increase 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 Division I is out-of-service (inoperable) for scheduled outage maintenance work and Division II is in-service (operable), then a loss of Vital DC power affecting Division II would require the declaration of an Unusual Event. A loss of Vital DC power to Division I 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 CA4, or an IC in Recognition Category R.

Susquehanna Basis Reference(s):

1. FSAR Section 8.3.2 DC Power Systems
2. Susquehanna Drawing No. E107159, Sheet 1, "Single Line Meter & Relay Diagram 125 VDC, 250 VDC & 120 VAC Systems"
3. Technical Specifications 3.8.5 DC Sources – Shutdown
4. ON-102(202)-610, -620, -630, -640 Loss of 125V DC
5. AR-1(2)06-001 Main Turbine/Generator, Computer HVAC, Instrument AC, 24V DC, 125V DC, 250V DC Panel 2C651
6. NEI 99-01 CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – RCS Temperature

Initiating Condition: Inability to maintain the plant in cold shutdown

EAL:

CA4.1 Alert

UNPLANNED increase in RCS temperature to > 200°F for > Table C-3 duration (Note 1)

OR

UNPLANNED RPV pressure increase > 10 psig due to loss of decay heat removal capability

Note 1: The ED/RM should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

Table C-3: RCS Heat-up Duration Thresholds

RCS Status	CONTAINMENT CLOSURE Status	Heat-up Duration
INTACT	N/A	60 min.*
Not INTACT	established	20 min.*
	not established	0 min.

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

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to Susquehanna, CONTAINMENT CLOSURE is established per NDAP-QA-0309 (ref. 3) for Primary Containment OR is established per NDAP-QA-0321 (ref. 4) for Secondary Containment.

RCS INTACT - The RCS should be considered intact when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals).

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.

Susquehanna Basis:

In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the RPV pressure increase criteria when the RCS is intact in Mode 4 or based on time to boil data when in Mode 5 or the RCS is not intact in Mode 4.

Available methods of determining RCS temperature can be found in Operations surveillance procedures (ref. 1, 2).

10 psig is the lowest pressure increase increment that can be reasonably read in the control room (ref. 2):

NEI 99-01 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.

The RCS Heat-up Duration Thresholds table addresses an increase in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not intact.. The 20-minute criterion was included to allow time for operator action to address the temperature increase.

The RCS Heat-up Duration Thresholds table also addresses an increase 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 increase without a substantial degradation in plant safety.

Finally, in the case where there is an increase in RCS temperature, the RCS is not intact , 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 Primary Containment or Reactor Building atmosphere and subsequently to the environment, and 2) there is reduced reactor coolant inventory above the top of irradiated fuel.

The RCS pressure increase threshold provides a pressure-based indication of RCS heat-up in the absence of RCS temperature monitoring capability.

Escalation of the emergency classification level would be via IC CS1 or RS1.

Susquehanna Basis Reference(s):

1. SO-100(200)-011 Reactor Vessel Temperature and Pressure Recording
2. SO-100(200)-006 Shiftly Surveillance Operating Log
3. NDAP-QA-0309 Primary Containment Access and Control
4. NDAP-QA-0321 Secondary Containment Integrity Control
5. NEI 99-01 CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU4.1	**Unusual Event**
UNPLANNED increase in RCS temperature to > 200°F due to loss of decay heat removal capability	

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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.

Susquehanna Basis:

In the absence of reliable RCS temperature indication caused by a loss of decay heat removal capability, classification should be based on EAL CU4.2 should RPV level indication be subsequently lost.

Available methods of determining RCS temperature can be found in Operations surveillance procedures (ref. 2, 3).

NEI 99-01 Basis:

This IC addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit and 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/Recovery Manager should also refer to IC CA4.

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

This EAL 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 at or 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 increase in reactor coolant temperature depending on the time after shutdown.

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA4 based on exceeding plant configuration-specific time criteria.

Susquehanna Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. SO-100(200)-011 Reactor Vessel Temperature and Pressure Recording
3. SO-100(200)-006 Shiftly Surveillance Operating Log
4. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU4.2	**Unusual Event**
Loss of **ALL** RCS temperature and RPV level indication for ≥ 15 min. (Note 1)	

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

None

Susquehanna Basis:

Available methods of determining RCS temperature can be found in Operations surveillance procedures (ref. 4, 5).

RPV level is normally monitored using the instruments in Figure C-2 (ref. 1, 2).

NEI 99-01 Basis:

This EAL addresses the inability to determine RCS temperature and RPV level, and 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/Recovery Manager should also refer to IC CA4.

This EAL 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.

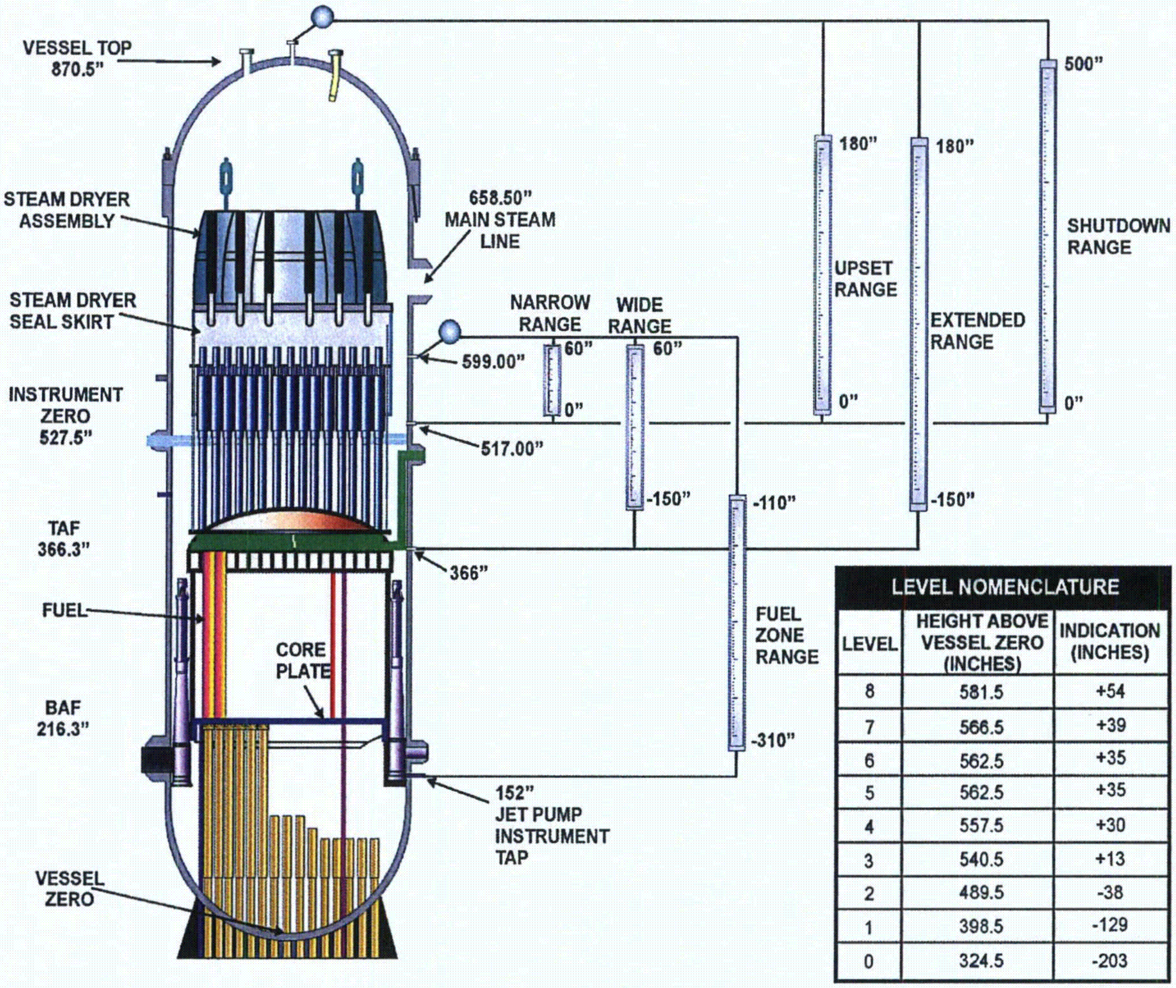
Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA4 based on exceeding plant configuration-specific time criteria.

Susquehanna Basis Reference(s):

1. M-142 P&ID Nuclear Boiler Vessel Instrumentation, Sheets 1, 2
2. ON-145(245)-004 RPV Water Level Anomaly
3. Technical Specifications Table 1.1-1
4. SO-100(200)-011 Reactor Vessel Temperature and Pressure Recording

5. SO-100(200)-006 Shiftly Surveillance Operating Log
6. NEI 99-01 CU3

Figure C-2 RPV Levels (ref. 1, 2)



LEVEL NOMENCLATURE		
LEVEL	HEIGHT ABOVE VESSEL ZERO (INCHES)	INDICATION (INCHES)
8	581.5	+54
7	566.5	+39
6	562.5	+35
5	562.5	+35
4	557.5	+30
3	540.5	+13
2	489.5	-38
1	398.5	-129
0	324.5	-203

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.1 Unusual Event

Loss of **ALL** Table C-4 onsite communication methods

OR

Loss of **ALL** Table C-4 ORO communication methods

OR

Loss of **ALL** Table C-4 NRC communication methods

Table C-4 Communication Methods

System	Onsite	ORO	NRC
UHF Radio	X		
Plant PA System	X		
Dedicated Conference Lines		X	
Commercial Telephone Systems	X	X	X
Cellular Telephone		X	X
FTS-2001 (ENS)		X	X

Mode Applicability:

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

Definition(s):

None

Susquehanna Basis:

Onsite/offsite communications include one or more of the systems listed in Table C-4 (ref. 1, 2, 3).

UHF Radio

Onsite portable radio communication systems are described in the Susquehanna SES Physical Security Plan and in the Susquehanna SES Emergency Plan. Four UHF channels, each consisting of two frequencies for duplex operation through one of five in-plant repeaters, provide onsite portable radio communications. Operations is assigned two channels; one channel is assigned to Unit 1 and one to Unit 2. Operators in the plant on rounds and on specific assignments are equipped with handheld two-way radios.

Plant PA System

The plant PA system is an intra-plant public address providing the following functions:

- A 5-channel page-talk handset intercom system for on-site communications between plant locations.
- Broadcast accountability and fire alarms designed to warn personnel of emergency conditions.

The system consists of telephone handsets, amplifiers and loudspeakers located at various selected areas throughout the plant.

Dedicated Conference Lines (Centrex Three (3) digit dialing)

The Dedicated Conference Lines are those normally used to communicate with several offsite agencies at one time (e.g., 191 conference line).

Commercial Telephone Systems

Two independent telecommunications networks exist to provide primary and backup telephone communications between ERFs and offsite agencies.

Plant Cellular Telephone

Cell phones can be utilized to perform both ORO and NRC communications.

FTS 2001 (ENS)

This system is for NRC offsite communications but may also be used to perform ORO notifications.

This EAL is the cold condition equivalent of the hot condition EAL SU7.1:

NEI 99-01 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.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the Commonwealth of Pennsylvania, Luzerne and Columbia County EOCs

The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

Susquehanna Basis Reference(s):

1. EP-RM-007 Emergency Telephone Instructions and Directory
2. SSES Emergency Plan Section 8
3. FSAR Section 9.5.2
4. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting Safety Systems

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

EAL:

CA6.1 Alert

The occurrence of **any** Table C-5 hazardous event

AND

EITHER:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Table C-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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 require a post-event inspection to determine if the attributes of an explosion are present.

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

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

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 (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

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.

Susquehanna Basis:

- The significance of a seismic event is discussed under EAL HU2.1 (ref. 1, 2).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 3, 4, 5).
- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 80 mph. (ref. 6, 7).
- Areas containing functions and systems required for safe shutdown of the plant are identified by Fire Zone in the fire response procedure (ref. 8, 9).
- An EXPLOSION that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

NEI 99-01 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.

The first conditional 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.

The second conditional 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.

Susquehanna Basis Reference(s):

1. ON-000-002 Severe Weather / Natural Phenomena
2. FSAR Section 3.7 Seismic Design
3. ON-169(269)-001 Flooding in Turbine Building
4. ON-169(269)-002 Flooding in Reactor Building
5. FSAR Section 3.4 Water Level (Flood) Design
6. FSAR Section 3.3 Wind and Tornado Loadings
7. FSAR Section 3.5 Missile Projection
8. SSES-FPRR Section 6.2 Fire Area Description
9. ON-013-001 Response to Fire
9. NEI 99-01 CA6

Category H – Hazards and Other Conditions Affecting Plant Safety

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

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

1. Security

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

2. Seismic Event

Natural events such as earthquakes have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety.

3. Natural or Technology Hazard

Other natural and non-naturally occurring events that can cause damage to plant facilities include tornados, FLOODING, hazardous material releases and events restricting site access warranting classification.

4. Fire

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

5. Hazardous Gas

Toxic, corrosive, asphyxiant or flammable gas leaks can affect normal plant operations or preclude access to plant areas required to safely shutdown the plant.

6. Control Room Evacuation

If the Control Room must be evacuated, additional support for monitoring and controlling plant functions is necessary through the emergency response facilities.

7. ED/RM Judgment

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

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HG1.1 General Emergency

A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision

AND

EITHER of the following has occurred:

Any of the following safety functions **cannot** be controlled or maintained

- Reactivity
- RPV water level
- RCS heat removal

OR

Damage to spent fuel has occurred or is IMMINENT

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward Susquehanna 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 Susquehanna. 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).

PROTECTED AREA - Area within the station inner security fence (PROTECTED AREA Barrier) designated to implement the requirements of 10 CFR 73.

Susquehanna Basis:

Security Shift Supervision are the designated on-site personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the SSES Physical Security Plan (Safeguards) information (ref. 1).

This IC does not apply to a HOSTILE ACTION directed at an ISFSI PROTECTED AREA; such an attack should be assessed using IC HA1.

If the plant equipment necessary to maintain the safety functions can be controlled from another location, then the EAL is not met.

Loss of SAS and/or CAS does not impact equipment needed for safety functions.

NEI 99-01 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*.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Susquehanna Physical Security Plan (ref.1).

Susquehanna Basis Reference(s):

1. SSES Physical Security Plan
2. ON-000-010 Security Event
3. NEI 99-01 HG1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA

EAL:

HS1.1	**Site Area Emergency**
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward Susquehanna 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 Susquehanna. 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).

PROTECTED AREA - Area within the station inner security fence (PROTECTED AREA Barrier) designated to implement the requirements of 10 CFR 73.

Susquehanna Basis:

Security Shift Supervision are the designated on-site personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the SSES Physical Security Plan (Safeguards) information (ref. 1).

NEI 99-01 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.*

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 Offsite Response Organization (ORO) resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

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

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Susquehanna Physical Security Plan (ref. 1).

Escalation of the emergency classification level would be via IC HG1.

Susquehanna Basis Reference(s):

1. SSES Physical Security Plan
2. ON-000-010 Security Event
3. NEI 99-01 HS1

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HA1.1	**Alert**
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervision	
OR	
A validated notification from NRC of an aircraft attack threat within 30 min. of the site	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward Susquehanna 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 Susquehanna. 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).

OWNER CONTROLLED AREA - Includes the area within the expanded security perimeter, i.e., the areas that are bordered by the Vehicle Barriers System. The OWNER CONTROLLED AREA also includes the Monitored OWNER CONTROLLED AREA (MOCA) as defined in Security Procedures.

Susquehanna Basis:

Security Shift Supervision are the designated on-site personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the SSES Physical Security Plan (Safeguards) information (ref. 1).

NEI 99-01 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 the 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*.

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 (OROs), 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.

The first threshold is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes any action directed against an ISFSI that is located outside the plant PROTECTED AREA.

The second threshold addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and OROs are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with site-specific security procedures.

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

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

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Susquehanna Physical Security Plan (ref. 1).

Susquehanna Basis Reference(s):

1. SSES Physical Security Plan
2. ON-000-010 Security Event
3. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1 Unusual Event

A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervision

OR

Notification of a credible security threat directed at the site

OR

A validated notification from the NRC providing information of an aircraft threat

Mode Applicability:

All

Definition(s):

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.

HOSTILE ACTION - An act toward Susquehanna 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 Susquehanna. 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).

Susquehanna Basis:

This EAL is based on the SSES Physical Security Plan (ref. 1).

Security Shift Supervision are the designated on-site personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the SSES Physical Security Plan (Safeguards) information (ref. 1).

NEI 99-01 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 Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

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*.

The first threshold references the Security Shift Supervision because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.39 information.

The second threshold addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the Susquehanna Physical Security Plan (ref. 1).

The third threshold 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 the Susquehanna Physical Security Plan (ref. 1).

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Susquehanna Physical Security Plan (ref. 1).

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

Susquehanna Basis Reference(s):

1. SSES Physical Security Plan
2. ON-000-010 Security Event
3. NEI 99-01 HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE levels

EAL:

HU2.1	**Unusual Event**
Seismic event greater than OPERATING BASIS EARTHQUAKE (OBE) as indicated by seismic instrumentation in the Control Room recording level greater than an OBE	

Mode Applicability:

All

Definition(s):

None

Susquehanna Basis:

Ground motion acceleration of 0.05g is the OPERATING BASIS EARTHQUAKE (OBE) for Susquehanna (ref. 1). Earthquake Monitoring Panel 0C696 provides indication of strong motion, OBE, or SAFE SHUTDOWN EARTHQUAKE (SSE) events for the Unit 1 Containment Foundation, Unit 2 Containment Foundation, and the ESW Pumphouse (as well as tape printout). Input from all six channels are recorded when a trigger initiates the system. A seismic event generally starts with an indication in the Control Room, Annunciator SEISMIC MON SYSTEM TRIGGERED (AR-016-001 window G06) on 0C653. The OBE is signaled by an LED illuminated green on the upper panel adjacent to the label, OPERATING BASIS EARTHQUAKE (ref. 2-4).

If above seismic monitoring is inoperable, ensure on-shift Operations and Security crews are aware of the need to report vibratory ground motion (earthquakes) to the control room, and the need for control room staff to contact outside agencies to confirm level of earthquake per OP-099-002 and ON-000-002.

To avoid inappropriate emergency classification resulting from spurious actuation of the seismic instrumentation or felt motion not attributable to seismic activity, an offsite agency (USGS, National Earthquake Information Center) can confirm that an earthquake has occurred in the area of the plant. Such confirmation should not, however, preclude a timely emergency declaration. The NEIC can be contacted by calling **(303) 273-8500**. Select **option #1** and inform the analyst you wish to confirm recent seismic activity in the vicinity of Susquehanna. Provide the analyst with the following Susquehanna coordinates: **41°5'20" north latitude, 76°8'56" west longitude** (ref. 4). Alternatively, near real-time seismic activity can be accessed via the NEIC website: *http://earthquake.usgs.gov/eqcenter/*

NEI 99-01 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.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event (e.g., lateral accelerations in excess of 0.08g). The Shift Manager or Emergency Director/Recovery Manager 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.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

Susquehanna Basis Reference(s):

1. TRM 3.3.2 Seismic Monitoring Instrumentation
2. FSAR Section 3.7a Seismic Design
3. ON-000-002 Natural Phenomena
4. OP-099-002 Seismic Monitoring System
5. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.1	**Unusual Event**
A tornado strike within the PROTECTED AREA	

Mode Applicability:

All

Definition(s):

PROTECTED AREA - Area within the station inner security fence (PROTECTED AREA Barrier) designated to implement the requirements of 10 CFR 73.

Susquehanna Basis:

If damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under EAL CA6.1 or SA8.1.

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

NEI 99-01 Basis:

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

EAL HU3.1 addresses a tornado striking (touching down) within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

Susquehanna Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.2	**Unusual Event**
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	

Mode Applicability:

All

Definition(s):

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

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 (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Susquehanna Basis:

Refer to NPE 91 001, SSES IPE, to identify susceptible internal flooding areas (ref. 1).

If the SAFETY SYSTEM component was operating at the time of isolation, EAL SA8.1 or CA6.1 may be applicable based on degraded SAFETY SYSTEM performance.

This EAL addresses water entering a room or area faster than installed equipment is capable of removing but is not applicable to water spraying on equipment of a magnitude that does not meet the definition of FLOODING.

NEI 99-01 Basis:

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

This EAL 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 caused by the FLOODING. Classification is also required if the water level or related wetting caused by the FLOODING 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.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

Susquehanna Basis Reference(s):

1. NPE 91 001 SSES IPE
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.3	**Unusual Event**
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)	

Mode Applicability:

All

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective measures such as temporary shielding, SCBAs or beyond Emergency Plan RWP dose extensions that are not routinely employed to access the room/area).

PROTECTED AREA - Area within the station inner security fence (PROTECTED AREA Barrier) designated to implement the requirements of 10 CFR 73.

Susquehanna Basis:

As used here, the term "offsite" is meant to be areas external to the PROTECTED AREA.

NEI 99-01 Basis:

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

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

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

Susquehanna Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.4	**Unusual Event**
A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)	

Note 7: This EAL does **not** apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Mode Applicability:

All

Definition(s):

None

Susquehanna Basis:

Events to consider are river flooding, hurricane, wind storms that block all of the multiple routes to get to the site.

NEI 99-01 Basis:

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

This EAL 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.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

Susquehanna Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.1 Unusual Event

A FIRE is not extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):

- 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

The FIRE is located within **any** Table H-1 area

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Control Structure • Diesel Generator Buildings • ESSW Pump House • Reactor Buildings • Turbine Buildings • ISFSI

Mode Applicability:

All

Definition(s):

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.

Susquehanna Basis:

The 15 minute requirement begins with a credible notification that a fire is occurring, or receipt of multiple valid fire detection system alarms or field validation of a single fire alarm. A single alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field. If a fire is verified to be occurring by field report, the 15 minute time limit is from the original receipt of multiple fire detection alarms/indications or field confirmation of the single fire detection alarm.

A valid fire detection alarm and a valid fire suppression alarm in the same area are considered receipt of multiple fire alarms or indications. Independent fire detection or suppression alarms in Table H-1 areas that are in close proximity are treated as multiple indications of a fire.

Table H-1 Fire Areas are based on the SSES Fire Protection Review Report (FPRR). Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS). (ref. 1)

NEI 99-01 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.

For EAL HU4.1 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 knowledge that a fires exists starts at the time of a report from the field, receipt of multiple fire detection alarms or indications, or field confirmation of a single alarm or indication. The fire duration clock starts at the time that knowledge of a fire exists.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

Susquehanna Basis Reference(s):

1. SSES FPRR Section 6.2 Fire Area Description
2. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.2 Unusual Event

Receipt of a single fire alarm (i.e., no other indications of a FIRE)

AND

The fire alarm is indicating a FIRE within **any** Table H-1 area

AND

The existence of a FIRE is not verified (i.e., proved or disproved) within 30 min. of alarm receipt (Note 1)

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Control Structure • Diesel Generator Buildings • ESSW Pump House • Reactor Buildings • Turbine Buildings • ISFSI

Mode Applicability:

All

Definition(s):

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.

Susquehanna Basis:

The 30 minute requirement begins upon receipt of a single valid fire detection or fire suppression system alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field. Actual field reports must be made within the 30 minute time limit or a classification must be made. If a fire is verified to be occurring by field report within 30 minutes, classification shall be made based on EAL HU4.1.

Table H-1 Fire Areas are based on the SSES Fire Protection Review Report (FPRR). Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS). (ref. 1)

NEI 99-01 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.

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 HU4.1 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.

<u>Basis-Related Requirements from Appendix R</u>

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 HU4.2, 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 SA8.

Susquehanna Basis Reference(s):

1. SSES FPRR Section 6.2 Fire Area Description
2. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.3	**Unusual Event**
A FIRE within the plant PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)	

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

All

Definition(s):

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.

PROTECTED AREA - Area within the station inner security fence (PROTECTED AREA Barrier) designated to implement the requirements of 10 CFR 73.

Susquehanna Basis:

None

NEI 99-01 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.

In addition to a FIRE addressed by EAL HU4.1 or HU4.2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety. Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

Susquehanna Basis Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.4	**Unusual Event**
A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish	

Mode Applicability:

All

Definition(s):

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.

PROTECTED AREA - Area within the station inner security fence (PROTECTED AREA Barrier) designated to implement the requirements of 10 CFR 73.

Susquehanna Basis:

None

NEI 99-01 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.

If a FIRE within the plant 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.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

Susquehanna Basis Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gases

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

EAL:

HA5.1 Alert

Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 rooms or areas

AND

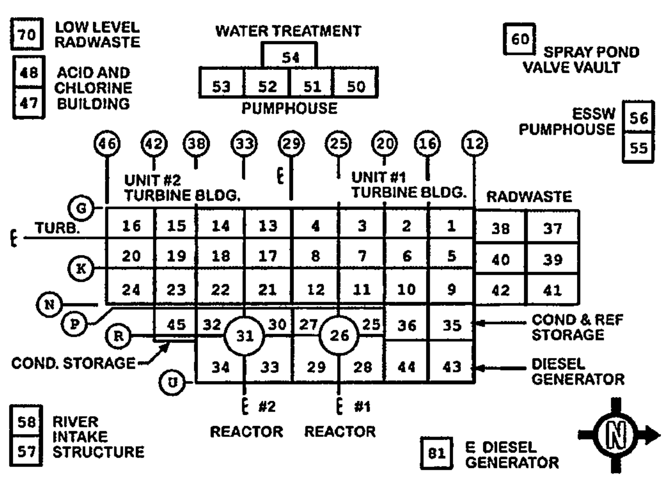
Entry into the room or area is prohibited or IMPEDED (Note 5)

Note 5: 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.

Table H-2 Safe Operation & Shutdown Areas			
Elevation	**Unit 1 Area(s) ****	**Unit 2 Area(s)****	**Mode(s)**
670'	RB 27	RB 32	3,4,5
683'	RB 27, 28, 29	RB 32, 33, 34	3,4,5
703'	RB 28, 29	RB 33, 34	3,4,5
719'	RB 25, 29	RB 30, 34	3,4,5
749'	RB 25, 29	RB 32, 33	3,4,5
729'	CS 12, 21	CS 12, 21	1, 2, 3, 4, 5, D

**** See Chart 1 for location of plant areas**

Chart 1- Plant Area Key Plan



Mode Applicability:

All

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., *requiring use of protective measures such as temporary shielding, SCBAs or beyond* Emergency Plan RWP dose extensions that are not routinely employed to access the room/area).

Susquehanna Basis:

If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

The list of plant areas in Table H-2 specify those areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area. See Chart 1 for the specific locations of areas listed in Table R-2. See Attachment 3 for more details of how the Table R-2 was developed (ref. 1).

NEI 99-01 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/Recover Manager'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.
- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the IDLH level of around 19.5%, 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, or to intentional inerting of containment.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

Susquehanna Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases
2. NEI 99-01 HA5

Category: *H – Hazards and Other Conditions Affecting Plant Safety*

Subcategory: 6 – Control Room Evacuation

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

EAL:

HS6.1 Site Area Emergency

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels

AND

Control of **any** of the following key safety functions is not reestablished within 15 min. (Note 1):

- Reactivity
- RPV water level
- RCS heat removal

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

All

Definition(s):

None

Susquehanna Basis:

The Shift Manager determines if the Control Room is uninhabitable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (Ref. 1, 2).

NEI 99-01 Basis:

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

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on Emergency Director/Recovery Manager judgment. The Emergency Director/Recovery Manager 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

Susquehanna Basis Reference(s):

1. ON-100(200)-009 Control Room Evacuation
2. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

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

EAL:

HA6.1	**Alert**
An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels	

Mode Applicability:

All

Definition(s):

None

Susquehanna Basis:

The Shift Manager (SM) determines if the Control Room is uninhabitable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (Ref. 1, 2).

Inability to establish plant control from outside the Control Room escalates this event to a Site Area Emergency per EAL HS6.1.

NEI 99-01 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.

Susquehanna Basis Reference(s):

1. ON-100(200)-009 Control Room Evacuation
2. NEI 99-01 HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – ED/RM Judgment

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

EAL:

HG7.1	**General Emergency**

Other conditions exist which in the judgment of the Emergency Director/Recovery Manager 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

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward Susquehanna 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 Susquehanna. 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).

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.

As applied to Susquehanna, CONTAINMENT CLOSURE is established per NDAP-QA-0309 (ref. 2) for Primary Containment OR is established per NDAP-QA-0321 (ref. 3) for Secondary Containment.

Susquehanna Basis:

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

NEI 99-01 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/Recovery Manager to fall under the emergency classification level description for a General Emergency.

Susquehanna Basis Reference(s):

1. Susquehanna LLC, Susquehanna Steam Electric Station Emergency Plan, Section 6.0 Organizational Control of Emergencies
2. NDAP-QA-0309 Primary Containment Access and Control
3. NDAP-QA-0321 Secondary Containment Integrity Control
4. NEI 99-01 HG7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – ED/RM Judgment

Initiating Condition: Other conditions existing that in the judgment of the Emergency Director/Recovery Manager warrant declaration of a Site Area Emergency

EAL:

HS7.1	**Site Area Emergency**
Other conditions exist which in the judgment of the Emergency Director/Recovery Manager 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 EMERGENCY PLAN BOUNDARY	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward Susquehanna 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 Susquehanna. 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).

EMERGENCY PLAN BOUNDARY (EPB) - Same as the Exclusion Area Boundary, i.e., that area around SSES within a radius of 1800 feet determined in accordance with 10 CFR 100.11.

Susquehanna Basis:

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

NEI 99-01 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/Recovery Manager to fall under the emergency classification level description for a Site Area Emergency.

Susquehanna Basis Reference(s):

1. Susquehanna LLC, Susquehanna Steam Electric Station Emergency Plan, Section 6.0 Organizational Control of Emergencies
2. NEI 99-01 HS7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – ED/RM Judgment

Initiating Condition: Other conditions exist that in the judgment of the Emergency Director/Recovery Manager warrant declaration of an Alert

EAL:

HA7.1	**Alert**
Other conditions exist which, in the judgment of the Emergency Director/Recovery Manager, 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.	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward Susquehanna 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 Susquehanna. 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).

Susquehanna Basis:

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

NEI 99-01 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/Recovery Manager to fall under the emergency classification level description for an Alert.

Susquehanna Basis Reference(s):

1. Susquehanna LLC, Susquehanna Steam Electric Station Emergency Plan, Section 6.0 Organizational Control of Emergencies
2. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – ED/RM Judgment

Initiating Condition: Other conditions existing that in the judgment of the Emergency Director/Recovery Manager warrant declaration of a UE

EAL:

HU7.1	**Unusual Event**
Other conditions exist which in the judgment of the Emergency Director/Recovery Manager 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.	

Mode Applicability:

All

Definition(s):

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 (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could *result in potential offsite exposures.*

Susquehanna Basis:

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

NEI 99-01 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/Recovery Manager to fall under the emergency classification level description for an Unusual Event.

Susquehanna Basis Reference(s):

1. Susquehanna LLC, Susquehanna Steam Electric Station Emergency Plan, Section 6.0 Organizational Control of Emergencies
2. NEI 99-01 HU7

Category S – System Malfunction

EAL Group: Hot Conditions (RCS temperature > 200°F); EALs in this category are applicable only in one or more hot operating modes.

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

The events of this category pertain to the following subcategories:

1. Loss of Emergency AC Power

Loss of emergency electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite sources for 4.16 kV ESS buses.

2. Loss of Vital DC Power

Loss of emergency electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of power to or degraded voltage on the 125 VDC vital buses.

3. Loss of Control Room Indications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

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

5. RCS Leakage

The reactor pressure vessel provides a volume for the coolant that covers the reactor core. The reactor pressure vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail. Excessive RCS leakage greater than Technical Specification limits indicates potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and Primary Containment integrity.

6. RPS Failure

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

7. Loss of Communications

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

8. Hazardous Event Affecting Safety Systems

Certain hazardous natural and technological events may result in visible damage to or degraded performance of safety systems warranting classification.

Category: S –System Malfunction

Subcategory: 1 – Loss of Essential AC Power

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

EAL:

SG1.1 General Emergency

Loss of **ALL** offsite and **ALL** onsite AC power capability to **ALL** 4.16 kV ESS buses on **EITHER** unit

AND

EITHER:

Restoration of at least one 4.16 kV ESS bus in < 4 hours is **not** likely (Note 1)

OR

RPV water level **CANNOT** BE RESTORED AND MAINTAINED > -179 in.

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

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

Definition(s):

None

Susquehanna Basis:

The Class 1E 4.16 kV system supplies all the Engineered Safety Feature (ESF) loads and other loads that are needed for a safe and orderly plant shutdown, and for keeping the plant in a safe shutdown condition. The eight Class 1E 4.16 kV ESS Buses 1(2)A through 1(2)D receive power from either the four ESS 13.8/4.16 kV transformers or the diesel generators (A, B, C, D and additional diesel generator E). Buses 1A-1D supply Unit 1 and common loads and Buses 2A-2D supply Unit 2 loads. This configuration prevents a loss of all ESS Buses for one unit in the event one of the ESS Transformers is lost.

During normal plant operation, ESS Transformer 101 supplies preferred power to ESS Bus 1A and 2A and is an alternate power supply to ESS bus 1D and 2D. ESS Transformer 111 supplies preferred power to ESS Bus 1C and 2C, and is an alternate power supply to ESS bus 1B and 2B. ESS Transformer 201 supplies preferred power to ESS Bus 1D and 2D, and is an alternate power supply to ESS bus 1A and 2A. ESS Transformer 211 supplies preferred power to ESS Bus 1B and 2B, and is an alternate power supply to ESS bus 1C and 2C.

On a loss of a preferred power source, the bus rapidly transfers to the alternate power source to maintain component power. If both the preferred and alternate power sources are lost, the associated standby diesel generator connects to the ESS bus. (ref. 2-6)

Four hours is the station blackout coping time (ref. 7).

Indication of continuing core cooling degradation is manifested by the inability to restore and maintain RPV water level above the Minimum Steam Cooling Reactor Water Level (-179 in.) (ref. 8). Core submergence is the most desirable means of core cooling, however when RPV level is below TAF, the uncovered portion of the core can be cooled by less reliable means (i.e., steam cooling or spray cooling).

NEI 99-01 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 the analyzed station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased 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.

Susquehanna Basis Reference(s):

1. FSAR Section 8.2 Offsite Power System
2. FSAR Section 8.3 Onsite Power System
3. Technical Specifications 3.8.1 AC Sources – Operating
4. Technical Specifications 3.8.7 Distribution System – Operating
5. ON-104 (204)-001 Units 1(2) Response to Loss of All Offsite Power
6. EO-100 (200)-030 UNIT 1(2) Response to Station Blackout
7. FSAR Section 15.9 STATION BLACKOUT (SBO)
8. EO-000-102 RPV Control
9. NEI 99-01 SG1

Category: S –System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of **all** essential AC and vital DC power sources for 15 minutes or longer

EAL:

SG1.2 General Emergency

Loss of **ALL** offsite and **ALL** onsite AC power capability to **ALL** 4.16 kV ESS buses on **EITHER** unit
for ≥ 15 min.

AND

Indicated voltage is < 105 VDC on **ALL** of the following vital 125 VDC main distribution buses on the affected unit for ≥ 15 min. (Note 1):

- 1D612 (2D612)
- 1D622 (2D622)
- 1D632 (2D632)
- 1D642 (2D642)

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

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

Definition(s):

None

Susquehanna Basis:

This EAL addresses operating experience from the March 2011 accident at Fukushima Daiichi.

The Class 1E 4.16 kV system supplies all the Engineered Safety Feature (ESF) loads and other loads that are needed for a safe and orderly plant shutdown, and for keeping the plant in a safe shutdown condition. The eight Class 1E 4.16 kV ESS Buses 1(2)A through 1(2)D receive power from either the four ESS 13.8/4.16 kV transformers or the diesel generators (A, B, C, D and additional diesel generator E). Buses 1A-1D supply Unit 1 and common loads and Buses 2A-2D supply Unit 2 loads. This configuration prevents a loss of all ESS Buses for one unit in the event one of the ESS Transformers is lost.

During normal plant operation, ESS Transformer 101 supplies preferred power to ESS Bus 1A and 2A and is an alternate power supply to ESS bus 1D and 2D. ESS Transformer 111 supplies preferred power to ESS Bus 1C and 2C, and is an alternate power supply to ESS bus 1B and 2B. ESS Transformer 201 supplies preferred power to ESS Bus 1D and 2D, and is an alternate power supply to ESS bus 1A and 2A. ESS Transformer 211 supplies preferred power to ESS Bus 1B and 2B, and is an alternate power supply to ESS bus 1C and 2C.

On a loss of a preferred power source, the bus rapidly transfers to the alternate power source to maintain component power. If both the preferred and alternate power sources are lost, the associated standby diesel generator connects to the ESS bus. (ref. 1-6)

The Class 1E Battery Banks are 1(2)D610 (Channel A), 1(2)D620 (Channel B), 1(2)D630 (Channel C), and 1(2)D640 (Channel D). Each bank consists of 60 cells connected in series. Each cell produces a nominal voltage of 2.06 VDC resulting in a total battery bank terminal voltage of 123.6 VDC. All battery banks are designed to supply power to its load center for four hours in the event of a loss of power from its battery charger (ref. 7-9).

105 VDC is the minimum design voltage limit (ref. 10).

Indicated voltage for the vital 125 VDC main distribution buses is local only. Local voltage indication is available for each bus based on dispatching a field operator in accordance with Control Room alarm response procedure AR-1(2)06-001 (A12,B12,C12,D12). Field observation of indicated voltage constitutes the point in time when availability of indications to plant operators that an emergency action level has been, or may be, exceeded.

NEI 99-01 Basis:

This IC addresses a concurrent and prolonged loss of both emergency AC and Vital DC power. A loss of all emergency 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 emergency AC and vital 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.

Susquehanna Basis Reference(s):

1. FSAR Section 8.2 Offsite Power System
2. FSAR Section 8.3 Onsite Power System
3. Technical Specifications 3.8.1 AC Sources – Operating
4. Technical Specifications 3.8.7 Distribution System – Operating
5. ON-104 (204)-001 Units 1(2) Response to Loss of All Offsite Power
6. EO-100 (200)-030 UNIT 1(2) Response to Station Blackout
7. FSAR Section 8.3.2 DC Power Systems
8. Susquehanna Drawing No. E107159, Sheet 1, "Single Line Meter & Relay Diagram 125 VDC, 250 VDC & 120 VAC Systems"
9. Technical Specifications 3.8.5 DC Sources – Shutdown
10. ON-102(202)-610, -620, -630, -640 Loss of 125V DC
11. AR-1(2)06-001 Main Turbine/Generator, Computer HVAC, Instrument AC, 24V DC, 125V DC, 250V DC Panel 2C651
12. NEI 99-01 SG8

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

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

EAL:

SS1.1	**Site Area Emergency**
Loss of **ALL** offsite and **ALL** onsite AC power capability to **ALL** 4.16 kV ESS buses on **EITHER** unit for ≥ 15 min. (Note 1)	

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

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

Definition(s):

None

Susquehanna Basis:

The Class 1E 4.16 kV system supplies all the Engineered Safety Feature (ESF) loads and other loads that are needed for a safe and orderly plant shutdown, and for keeping the plant in a safe shutdown condition. See Figure S-1 (ref. 1, 2) The eight Class 1E 4.16 kV ESS Buses 1(2)A through 1(2)D receive power from either the four ESS 13.8/4.16 kV transformers or the diesel generators (A, B, C, D and additional diesel generator E). Buses 1A-1D supply Unit 1 and common loads and Buses 2A-2D supply Unit 2 loads. This configuration prevents a loss of all ESS Buses for one unit in the event one of the ESS Transformers is lost.

During normal plant operation, ESS Transformer 101 supplies preferred power to ESS Bus 1A and 2A and is an alternate power supply to ESS bus 1D and 2D. ESS Transformer 111 supplies preferred power to ESS Bus 1C and 2C, and is an alternate power supply to ESS bus 1B and 2B. ESS Transformer 201 supplies preferred power to ESS Bus 1D and 2D, and is an alternate power supply to ESS bus 1A and 2A. ESS Transformer 211 supplies preferred power to ESS Bus 1B and 2B, and is an alternate power supply to ESS bus 1C and 2C.

On a loss of a preferred power source, the bus rapidly transfers to the alternate power source to maintain component power. If both the preferred and alternate power sources are lost, the associated standby diesel generator connects to the ESS bus. (ref. 2-6)

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. The interval begins when both offsite and onsite AC power are lost.

NEI 99-01 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.

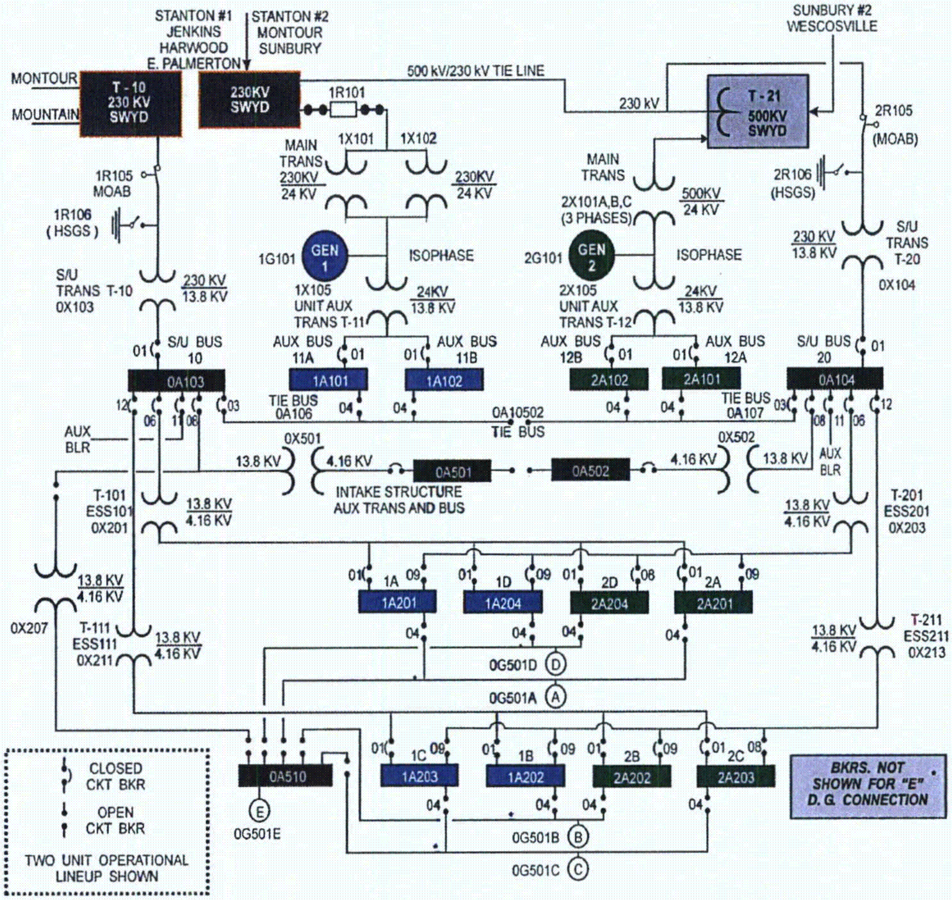
Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

Susquehanna Basis Reference(s):

1. FSAR Section 8.2 Offsite Power System
2. FSAR Section 8.3 Onsite Power System
3. Technical Specifications 3.8.1 AC Sources – Operating
4. Technical Specifications 3.8.7 Distribution System – Operating
5. ON-104 (204)-001 Units 1(2) Response to Loss of All Offsite Power
6. EO-100 (200)-030 UNIT 1(2) Response to Station Blackout
7. NEI 99-01 SS1

Figure S-1 ESS 13.8/4.16 kV Transformers and Distribution (ref. 1)



Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of **all but one** AC power source to essential buses for 15 minutes or longer

EAL:

SA1.1 **Alert**
AC power capability to **ALL** 4.16 kV ESS buses on **EITHER** unit reduced to a single power source for ≥ 15 min. (Note 1)
AND
Any additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

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

Definition(s):

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 (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

Susquehanna Basis:

The Class 1E 4.16 kV system supplies all the Engineered Safety Feature (ESF) loads and other loads that are needed for a safe and orderly plant shutdown, and for keeping the plant in a safe shutdown condition. See Figure S-1 (ref. 1, 2) The eight Class 1E 4.16 kV ESS Buses 1(2)A through 1(2)D receive power from either the four ESS 13.8/4.16 kV transformers or the diesel generators (A, B, C, D and additional diesel generator E). Buses 1A-1D supply Unit 1 and common loads and Buses 2A-2D supply Unit 2 loads. This configuration prevents a loss of all ESS Buses for one unit in the event one of the ESS Transformers is lost.

During normal plant operation, ESS Transformer 101 supplies preferred power to ESS Bus 1A and 2A and is an alternate power supply to ESS bus 1D and 2D. ESS Transformer 111 supplies preferred power to ESS Bus 1C and 2C, and is an alternate power supply to ESS bus

1B and 2B. ESS Transformer 201 supplies preferred power to ESS Bus 1D and 2D, and is an alternate power supply to ESS bus 1A and 2A. ESS Transformer 211 supplies preferred power to ESS Bus 1B and 2B, and is an alternate power supply to ESS bus 1C and 2C.

On a loss of a preferred power source, the bus rapidly transfers to the alternate power source to maintain component power. If both the preferred and alternate power sources are lost, the associated standby diesel generator connects to the ESS bus. (ref. 2-6)

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

This hot condition EAL is equivalent to the cold condition EAL CU2.1.

NEI 99-01 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 capable of supplying required power to an emergency bus. Some examples of this condition are presented below.

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

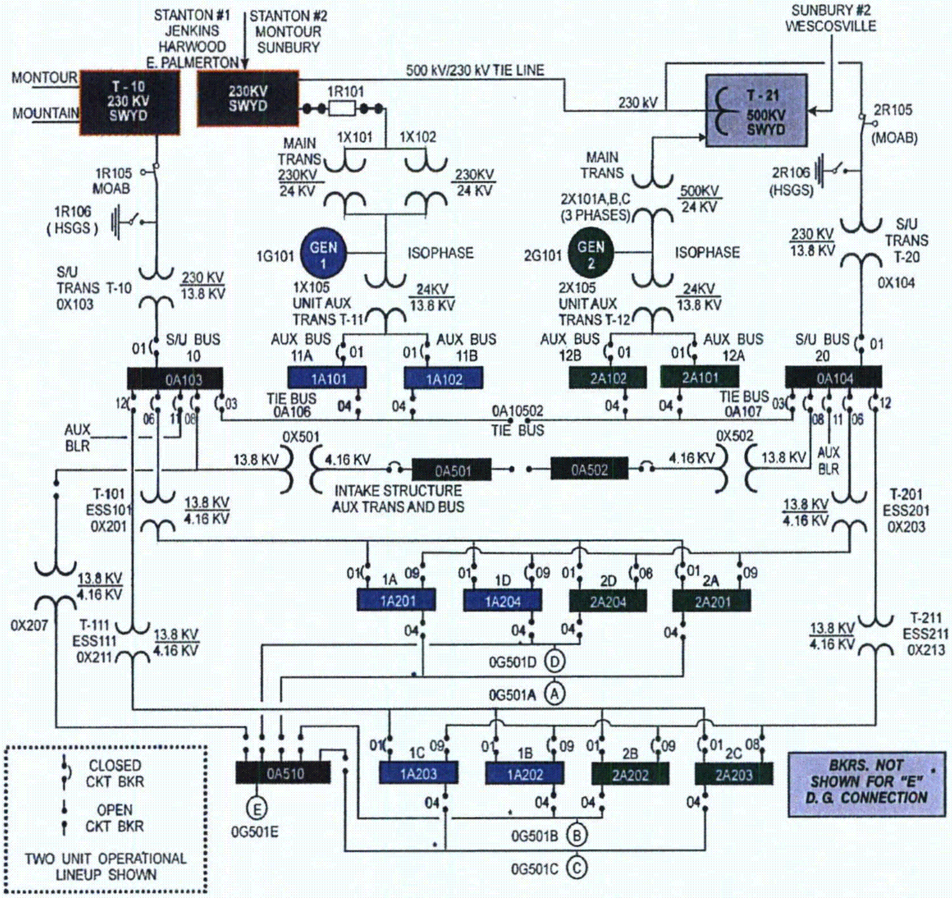
Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

Escalation of the emergency classification level would be via IC SS1.

Susquehanna Basis Reference(s):

1. FSAR Section 8.2 Offsite Power System
2. FSAR Section 8.3 Onsite Power System
3. Technical Specifications 3.8.1 AC Sources – Operating
4. Technical Specifications 3.8.7 Distribution System – Operating
5. ON-104 (204)-001 Units 1(2) Response to Loss of All Offsite Power
6. EO-100 (200)-030 UNIT 1(2) Response to Station Blackout
7. NEI 99-01 SA1

Figure S-1 ESS 13.8/4.16 kV Transformers and Distribution (ref. 1)



Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

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

EAL:

SU1.1	**Unusual Event**
Loss of **ALL** offsite AC power capability to **ALL** 4.16 kV ESS buses on **EITHER** unit for ≥ 15 min. (Note 1)	

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

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

Definition(s):

None

Basis:

Susquehanna Basis:

The Class 1E 4.16 kV system supplies all the Engineered Safety Feature (ESF) loads and other loads that are needed for a safe and orderly plant shutdown, and for keeping the plant in a safe shutdown condition. See Figure S-1 (ref. 1, 2) The eight Class 1E 4.16 kV ESS Buses 1(2)A through 1(2)D receive power from either the four ESS 13.8/4.16 kV transformers or the diesel generators (A, B, C, D and additional diesel generator E). Buses 1A-1D supply Unit 1 and common loads and Buses 2A-2D supply Unit 2 loads. This configuration prevents a loss of all ESS Buses for one unit in the event one of the ESS Transformers is lost.

During normal plant operation, ESS Transformer 101 supplies preferred power to ESS Bus 1A and 2A and is an alternate power supply to ESS bus 1D and 2D. ESS Transformer 111 supplies preferred power to ESS Bus 1C and 2C, and is an alternate power supply to ESS bus 1B and 2B. ESS Transformer 201 supplies preferred power to ESS Bus 1D and 2D, and is an alternate power supply to ESS bus 1A and 2A. ESS Transformer 211 supplies preferred power to ESS Bus 1B and 2B, and is an alternate power supply to ESS bus 1C and 2C.

On a loss of a preferred power source, the bus rapidly transfers to the alternate power source to maintain component power. If both the preferred and alternate power sources are lost, the associated standby diesel generator connects to the ESS bus. (ref. 2-6)

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses.

NEI 99-01 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.

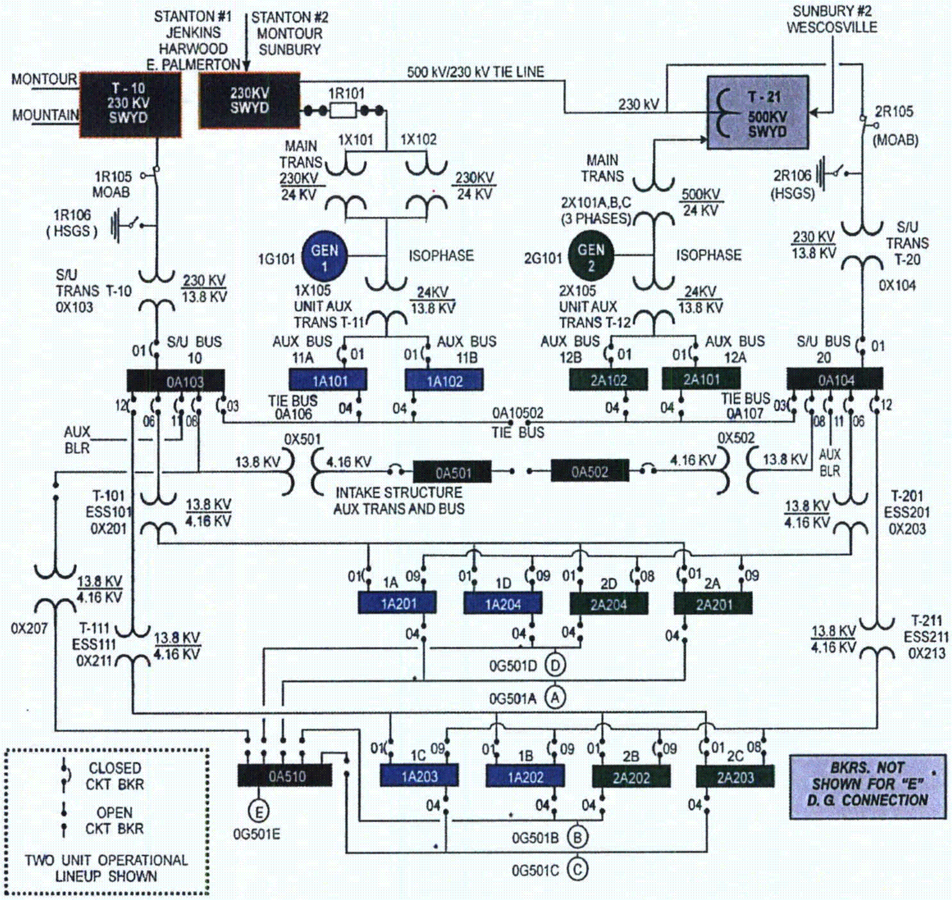
Fifteen minutes was selected as a threshold to exclude transient or momentary losses of offsite power.

Escalation of the emergency classification level would be via IC SA1.

Susquehanna Basis Reference(s):

1. FSAR Section 8.2 Offsite Power System
2. FSAR Section 8.3 Onsite Power System
3. Technical Specifications 3.8.1 AC Sources – Operating
4. Technical Specifications 3.8.7 Distribution System – Operating
5. ON-104 (204)-001 Units 1(2) Response to Loss of All Offsite Power
6. EO-100 (200)-030 UNIT 1(2) Response to Station Blackout
7. NEI 99-01 SU1

Figure S-1 ESS 13.8/4.16 kV Transformers and Distribution (ref. 1)



Category: S – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

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

EAL:

SS2.1 Site Area Emergency

Indicated voltage is < 105 VDC on **ALL** of the following vital 125 VDC main distribution buses on the affected unit for ≥ 15 min. (Note 1):

- 1D612 (2D612)
- 1D622 (2D622)
- 1D632 (2D632)
- 1D642 (2D642)

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

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

Definition(s):

None

Susquehanna Basis:

The Class 1E Battery Banks are 1(2)D610 (Channel A), 1(2)D620 (Channel B), 1(2)D630 (Channel C), and 1(2)D640 (Channel D). Each bank consists of 60 cells connected in series. Each cell produces a nominal voltage of 2.06 VDC resulting in a total battery bank terminal voltage of 123.6 VDC. All battery banks are designed to supply power to its load center for four hours in the event of a loss of power from its battery charger (ref. 1-3).

105 VDC is the minimum design voltage limit (ref. 4).

Indicated voltage for the vital 125 VDC main distribution buses is local only. Local voltage indication is available for each bus based on dispatching a field operator in accordance with Control Room alarm response procedure AR-1(2)06-001 (A12,B12,C12,D12). Field observation of indicated voltage constitutes the point in time when availability of indications to plant operators that an emergency action level has been, or may be, exceeded.

NEI 99-01 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 SG1.

Susquehanna Basis Reference(s):

1. FSAR Section 8.3.2 DC Power Systems
2. Susquehanna Drawing No. E107159, Sheet 1, "Single Line Meter & Relay Diagram 125 VDC, 250 VDC & 120 VAC Systems"
3. Technical Specifications 3.8.5 DC Sources – Shutdown
4. ON-102(202)-610, -620, -630, -640 Loss of 125V DC
5. AR-1(2)06-001 Main Turbine/Generator, Computer HVAC, Instrument AC, 24V DC, 125V DC, 250V DC Panel 2C651
6. NEI 99-01 SS8

Category:	S – System Malfunction
Subcategory:	3 – Loss of Control Room Indications
Initiating Condition:	UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress

EAL:

SA3.1 Alert

An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

Any significant transient is in progress, Table S-2

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 Safety System Parameters
• Reactor power • RPV water level • RPV pressure • Primary Containment pressure • Suppression Pool water level • Suppression Pool temperature

Table S-2 Significant Transients
• Reactor scram • Runback > 25% reactor power • RRC pump trip while > 25% reactor power • ECCS injection • Thermal power oscillations > 10%

Mode Applicability:

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

Definition(s):

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.

Susquehanna Basis:

SAFETY SYSTEM parameters listed in Table S-1 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Process Computer (PPC) and SPDS are redundant compensatory indication which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

Significant transients are listed in Table S-2 and include response to automatic or manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, RRC pump trip while > 25% reactor power, ECCS injections, or thermal power oscillations of 10% or greater.

NEI 99-01 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, RPV level 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 RPV water 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

Susquehanna Basis Reference(s):

1. FSAR Section 18.1.17 Plant Safety Parameter Display System
2. OP-131(231)-002 Plant Computer Systems

3. EO-000-102 RPV Control
4. EO-000-103 Primary Containment Control
5. NEI 99-01 SA2

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

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

EAL:

SU3.1	**Unusual Event**
An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1)	

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 Safety System Parameters
• Reactor power • RPV water level • RPV pressure • Primary Containment pressure • Suppression Pool water level • Suppression Pool temperature

Mode Applicability:

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

Definition(s):

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.

Susquehanna Basis:

SAFETY SYSTEM parameters listed in Table S-1 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Process Computer (PPC) and SPDS are redundant compensatory indication which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

NEI 99-01 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, RPV level 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 RPV water 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 SA3.

Susquehanna Basis Reference(s):

1. FSAR Section 18.1.17 Plant Safety Parameter Display System
2. OP-131(231)-002 Plant Computer Systems
3. EO-000-102 RPV Control
4. EO-000-103 Primary Containment Control
5. NEI 99-01 SU2

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

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

EAL:

SU4.1	Unusual Event
Offgas pretreatment monitor high-high radiation alarm	

Mode Applicability:

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

Definition(s):

None

Susquehanna Basis:

The Offgas Pretreatment RMS monitors radioactivity in the Offgas system downstream of the Motive Steam Jet Condenser. The monitor detects the radiation level that is attributable to the fission gases produced in the reactor and transported with steam through the turbine to the condenser (ref. 1). Log rad monitors and trip auxiliary units are located on Panel 1C604 in the Upper Relay Room. Instrument Channel 'A' is RITS-D12-1K601A and Instrument Channel 'B' is RITS-D12-1K601B. Both channels output to Yokagowa Recorder RR-D12-1R601 on Main Control Room Panel 1C600 (ref. 2, 3).

OFFGAS HI-HI RADIATION (AR-106-F03) is located on Panel 1C651. The setpoint is variable based on surveillance procedure (ref. 4).

NEI 99-01 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.

Susquehanna Basis Reference(s):

1. FSAR Section 11.5 Process and Effluent Radiological Monitoring and Sampling Systems
2. Technical Specification 3.7.5 Main Condenser Offgas
3. AR-106(206)-001 F03 Offgas Hi Hi Radiation
4. SC-143(243)-101 Unit 1 (Unit 2) Main Condenser Air Ejector Monthly Noble Gas Activity
5. OP-179(279)-002 Process Radiation Monitoring System
6. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

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

EAL:

SU4.2 Unusual Event

Coolant activity > 0.2 µCi/gm dose equivalent I-131 for > 48 hours

OR

Coolant activity > 4.0 µCi/gm dose equivalent I-131 at any time

Mode Applicability:

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

Definition(s):

None

Susquehanna Basis:

The specific iodine activity is limited to ≤ 0.2 µCi/gm dose equivalent I-131 (Condition A) with a completion time of 48 hours. This limit ensures the source term assumed in the safety analysis for the Main Steam Line Break (MSLB) is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the regulatory limits (ref. 1).

The upper limit of 4.0 µCi/gm dose equivalent I-131 (Condition B) ensures that the TEDE dose from an MSLB will not exceed the dose guidelines of 10 CFR 50.67 or Control Room operator dose limits specified in GDC 19 of 10 CFR 50, Appendix A (ref. 1).

NEI 99-01 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.

Susquehanna Basis Reference(s):

1. Technical Specifications section 3.4.7 RCS Specific Activity
2. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1 Unusual Event

RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min.

OR

RCS identified leakage > 25 gpm for ≥ 15 min.

OR

Leakage from the RCS to a location outside Primary Containment > 25 gpm for ≥ 15 min. (Note 1)

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

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

Definition(s):

None

Susquehanna Basis:

Leakage is monitored by utilizing the following techniques (ref. 1):

- Monitoring changes in water level in the drywell floor drain sumps and drywell equipment drain tank
- Sensing excess flow in piping systems
- Sensing pressure and temperature changes in the primary containment
- Monitoring for high flow and temperature through selected drains,
- Sampling airborne particulate and gaseous radioactivity.

Identified leakage is leakage into the drywell, such as that from pump seals or valve packing, that is captured and conducted to the drywell equipment drain tank; or leakage into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary

Unidentified leakage is all leakage into the drywell that is not identified leakage (ref. 2).

Pressure boundary leakage is leakage through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall (ref. 2).

Two drywell floor drain sumps are located in the primary containment for collection of leakage from vent coolers, control rod drive flange leakage, chilled water drains, cooling water drains, and overflow from the drywell equipment drain tank. The drywell floor drain sumps are located at the drywell diaphragm slab low point. Unidentified leakages will, by gravity, flow down the slab surface into the floor drain sumps. Water flow rate greater than 0.5 gpm can be detected by monitoring changes of level over a time period. The sump depth of 0-5 in. is displayed on a 0-100 percent recorder chart, which relates to the sump nominal capacity of 0-150 gal.

The drywell equipment drain tank collects identified leakage within the primary containment from reactor head seal leak off, bulkhead drain, refueling bellows drain, RPV head vent, recirculation pump seals, reactor recirculation pump cooler drains, and RPV bottom drain (Unit 1 only). The measured tank depth of 36 in. is displayed on a 0-100 percent recorder chart. This relates directly to the measured tank capacity of 842 gal.

RCS leakage outside of the Primary Containment that is not considered identified or unidentified leakage per Technical Specifications includes leakage via interfacing systems such as RCS to the Reactor Building Closed Cooling Water (RBCCW system), or systems that directly see RCS pressure outside primary containment such as Reactor Water Cleanup (RWCU), reactor water sampling system and Residual Heat Removal (RHR) system (when in the shutdown cooling mode) (ref. 1, 3).

Indicated changes in drywell sump water level are used to calculate unidentified drywell leakage. Indicated changes in drywell equipment drain tank level are used to calculate identified drywell leakage. SO-100-006 and SO-200-006 are the procedures that specify how to complete these calculations.

Drywell leakage calculations in SO-100(200)-006 take a finite period of time to complete. Leakage rates cannot be determined quickly by merely observing an indicator. For this reason, the 15 minutes clock starts after it is determined that leakage rates exceed the entry value. Upon determination that leakage has increased substantially, effort should be made to quantify this leakage in a timely manner.

ON-1(2)00-005, "Excessive Drywell Leakage Identification", contains methods of quickly estimating drywell leakage. These methods can be used in lieu of completing the calculations contained in SO-1(2)00-006.

Means to directly quantify RCS leakage outside containment may not be available. For this reason, judgment must be used for assessment of the 25 gpm leak rate criterion. For example, a short steam plume that does not appreciably change room temperature or room radiation levels can be judged to be less than 25 gpm. A leak that causes room temperature to rise rapidly above maximum safe temperatures could be judged to be greater than 25 gpm in the absence of measurable leak rates, and thus judgment is an acceptable method to evaluate this criterion.

Escalation of this EAL to the Alert level is via Category F, Fission Product Barrier Degradation, EAL FA1.1. The note has been added to remind the EAL-user to review Table F-1 for possible escalation to higher emergency classifications.

NEI 99-01 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.

The first and second EAL conditions 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). The third condition addresses an RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These conditions thus apply to leakage into the Primary Containment, or a location outside of Primary Containment.

The leak rate values for each condition 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). The first condition uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

A stuck-open Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.

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

Escalation of the emergency classification level would be via ICs of Recognition Category R or F.

Susquehanna Basis Reference(s):

1. FSAR Section 5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary
2. Technical Specifications Definitions Section 1.1
3. ON-100(200)-005 Excess Drywell Leakage Identification
4. SO-100(200)-006
5. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to RPV water level or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual scram fails to shut down the reactor

AND

ALL actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5%

AND

EITHER:

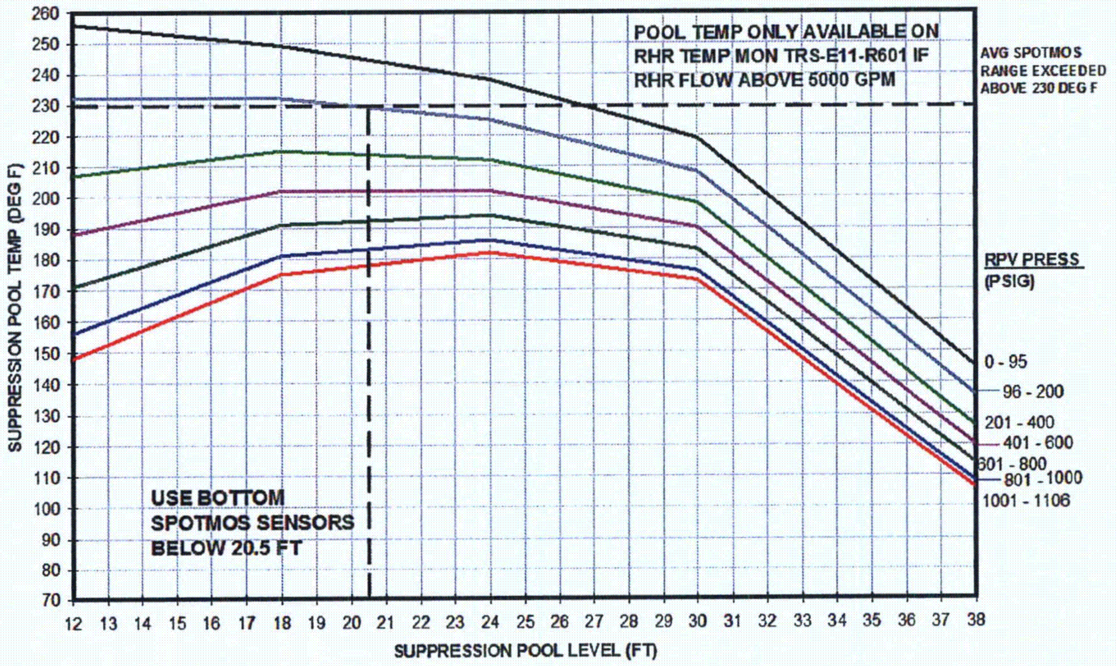
- RPV level **CANNOT** BE RESTORED AND MAINTAINED > -179 in. or **CANNOT** be determined

  OR

- Suppression pool water temperature AND RPV pressure **CANNOT** BE MAINTAINED below the Heat Capacity Temperature Limit (Figure - HCTL)

Figure – HCTL

Heat Capacity Temperature Limit



Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

Susquehanna Basis:

This EAL addresses the following:

- Any automatic reactor scram signal followed by a manual scram actions that fail to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed (EAL SA6.1), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

Reactor shutdown achieved by use of EO-000-113, Control Rod Insertion, is also credited as a successful scram provided reactor power can be reduced below the APRM downscale trip setpoint before indications of an extreme challenge to either core cooling or heat removal exist. (ref. 1)

The APRM downscale trip setpoint (5%) is a minimum reading on the power range scale that indicates power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, torus temperature trend) can be used to determine if reactor power is greater than 2% power (ref. 2, 3).

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

Indication that core cooling is extremely challenged is manifested by inability to restore and maintain RPV water level above the Minimum Steam Cooling RPV Water Level (MSCRWL) (ref. 3). The MSCRWL is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500°F. This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core spray cooling could be a precursor of a core melt sequence. When RPV level cannot be determined, EOPs require entry to EO-000-114, RPV Flooding. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EO-000-114 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (ref. 4).

The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool water temperature from which Emergency RPV Depressurization will not raise suppression chamber pressure above the Primary Containment Pressure Limit, while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

The HCTL is a function of RPV pressure and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant. This threshold is met when the final step of section SP/T in EO-000-103, Primary Containment Control, is reached (ref. 5). This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature.

NEI 99-01 Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram 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 shut down the reactor. The inclusion of this IC and EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shut down the reactor.

The adequacy of reactor shutdown (< 5%) is determined in accordance with applicable Emergency Operating Procedure criteria.

Escalation of the emergency classification level would be via IC RG1 or FG1.

Susquehanna Basis Reference(s):

1. EO-000-113 Control Rod Insertion
2. Technical Specifications Table 3.3.1.1-1
3. EO-000-102 RPV Control
4. EO-000-114 RPV Flooding
5. EO-000-103 Primary Containment Control
6. NEI 99-01 SS5

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor

EAL:

SA6.1	**Alert**
An automatic or manual scram fails to shut down the reactor	
AND	
Manual scram actions taken at the reactor control console (Manual PBs, Mode Switch, ARI) are **not** successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8)	

Note 8: A manual scram 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.

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

Susquehanna Basis:

This EAL addresses any automatic or manual reactor scram signal that fails to shut down the reactor followed by a subsequent manual scram that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed.

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., manual scram pushbuttons, mode switch, or ARI initiation in accordance with EO-000-102 or EO-000-113). Reactor shutdown achieved by use of other control rod insertion methods (e.g. individual control rod insertion) directed by EO-000-113 does not constitute a successful manual scram (ref. 2, 3).

For the purposes of this EAL, a successful <u>automatic</u> initiation of ARI that reduces reactor power below 5% is <u>not</u> considered a successful automatic scram. If automatic actuation of ARI has occurred and caused reactor shutdown, the automatic RPS scram must have failed. ARI is a backup means of inserting control rods in the unlikely event that an automatic RPS scram signal exists but the reactor continues to generate significant power. However, a successful automatic initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

The APRM downscale trip setpoint (5%) is a minimum reading on the power range scale that indicates power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, torus temperature trend) can be used to determine if reactor power is greater than 5% power (ref. 1, 3).

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

NEI 99-01 Basis:

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

A manual action at the reactor control console 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 scram). 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 console (e.g., locally opening breakers). Actions taken at backpanels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control console".

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram 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 shut down the reactor is prolonged enough to cause a challenge to RPV water level or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6 or FS1, an Alert declaration is appropriate for this event.

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.

The adequacy of reactor shutdown (< 5%) is determined in accordance with applicable Emergency Operating Procedure criteria.

Susquehanna Basis Reference(s):

1. Technical Specifications Table 3.3.1.1-1
2. EO-000-113 Control Rod Insertion
3. EO-000-102 RPV Control
4. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor

EAL:

SU6.1 Unusual Event

An automatic scram did **not** shut down the reactor after **any** RPS setpoint is exceeded

AND

A subsequent automatic scram or manual scram action taken at the reactor control console (Manual PBs, Mode Switch, ARI) is successful in shutting down the reactor as indicated by reactor power < 5% (APRM downscale) (Note 8)

Note 8: A *manual scram 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.

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

Susquehanna Basis:

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function. A reactor scram is automatically initiated by the Reactor Protection System (RPS) when certain continuously monitored parameters exceed predetermined setpoints (ref. 1).

Following a successful reactor scram, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative period. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-scram response from an automatic reactor scram signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful scram has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power below the APRM downscale setpoint of 5%.

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., manual scram pushbuttons, mode switch, or ARI initiation in accordance with EO-000-102 or EO-000-113). Reactor shutdown achieved by use of other control rod insertion methods (e.g. individual control rod insertion) directed by EO-000-113 does not constitute a successful manual scram (ref. 2, 3).

Following any automatic RPS scram signal, operating procedures (e.g., EO-000-102) prescribe insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved. Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Unusual Event. (ref. 3)

Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

For the purposes of this EAL, a successful automatic initiation of ARI that reduces reactor power below 5% is not considered a successful automatic scram. If automatic initiation of ARI has occurred and caused reactor shutdown, the automatic RPS scram must have failed. ARI is a backup means of inserting control rods in the unlikely event that an automatic RPS scram signal exists but the reactor continues to generate significant power. However, a successful automatic or manual initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

In the event that the operator identifies a reactor scram is imminent and initiates a successful manual reactor scram before the automatic scram setpoint is reached, no declaration is required. The successful manual scram of the reactor before it reaches its automatic scram setpoint or reactor scram signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. If manual reactor scram actions fail to reduce reactor power below 5%, the event escalates to the Alert under EAL SA6.1.

If by procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 10CFR 50.72 should be considered for the transient event.

NEI 99-01 Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram 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 scram, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram). 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 scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram) using a different switch). Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles 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 scram). 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 consoles".

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

The adequacy of reactor shutdown (< 5%) is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor scram 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 reactor scram 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 scram 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.

Susquehanna Basis Reference(s):

1. Technical Specifications Table 3.3.1.1-1
2. EO-000-113 Control Rod Insertion
3. EO-000-102 RPV Control
4. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor

EAL:

SU6.2 Unusual Event

A manual scram did **not** shut down the reactor after **any** manual scram action was initiated

AND

A subsequent automatic scram or manual scram action taken at the reactor control console (Manual PBs, Mode Switch, ARI) is successful in shutting down the reactor as indicated by reactor power < 5% (APRM downscale) (Note 8)

Note 8: A manual scram 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.

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

Susquehanna Basis:

This EAL addresses a failure of a manually initiated scram in the absence of having exceeded an automatic RPS trip setpoint and a subsequent automatic or manual scram is successful in shutting down the reactor (reactor power < 5%) (ref. 1).

Following a successful reactor scram, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative period. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-scram response from a manual reactor scram signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful scram has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power below the APRM downscale setpoint of 5%.

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., manual scram pushbuttons, mode switch, or ARI initiation in accordance with EO-000-102 or EO-000-113). Reactor shutdown achieved by use of other control rod insertion methods (e.g. individual control rod insertion) directed by EO-000-113 does not constitute a successful manual scram (ref. 2, 3).

Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

Successful automatic or manual initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

If both subsequent automatic and subsequent manual reactor scram actions in the Control Room fail to reduce reactor power below the power associated with the safety system design (< 5%) following a failure of an initial manual scram, the event escalates to an Alert under EAL SA6.1

NEI 99-01 Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram 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 scram, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram). 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 scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram) using a different switch). Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles 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 scram). 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 consoles".

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

The adequacy of reactor shutdown (≤ 5%) is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor scram 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 reactor scram 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 scram 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.

Susquehanna Basis Reference(s):

1. Technical Specifications Table 3.3.1.1-1
2. EO-000-113 Control Rod Insertion
3. EO-000-102 RPV Control
4. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

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

EAL:

SU7.1 Unusual Event

Loss of **ALL** Table S-3 onsite communication methods

OR

Loss of **ALL** Table S-3 ORO communication methods

OR

Loss of **ALL** Table S-3 NRC communication methods

Table S-3 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
UHF Radio	X		
Plant PA System	X		
Dedicated Conference Lines		X	
Commercial Telephone Systems	X	X	X
Cellular Telephone		X	X
FTS-2001 (ENS)		X	X

Mode Applicability:

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

Definition(s):

None

Susquehanna Basis:

Onsite/offsite communications include one or more of the systems listed in Table S-3 (ref. 1, 2, 3).

<u>UHF Radio</u>

Onsite portable radio communication systems are described in the Susquehanna SES Physical Security Plan and in the Susquehanna SES Emergency Plan. Four UHF channels, each consisting of two frequencies for duplex operation through one of five in-plant repeaters, provide onsite portable radio communications. Operations is assigned two channels; one channel is assigned to Unit 1 and one to Unit 2. Operators in the plant on rounds and on specific assignments are equipped with handheld two-way radios.

Plant PA System

The plant PA system is an intra-plant public address providing the following functions:

- A 5-channel page-talk handset intercom system for on-site communications between plant locations.
- Broadcast accountability and fire alarms designed to warn personnel of emergency conditions.

The system consists of telephone handsets, amplifiers and loudspeakers located at various selected areas throughout the plant.

Dedicated Conference Lines (Centrex Three (3) digit dialing)

The Dedicated Conference Lines are those normally used to communicate with several offsite agencies at one time (e.g., 191 conference line).

Commercial Telephone Systems

Two independent telecommunications networks exist to provide primary and backup telephone communications between ERFs and offsite agencies.

Cellular Telephone

Cell phones can be utilized to perform both ORO and NRC communications.

FTS 2001 (ENS)

This system is for NRC offsite communications but may also be used to perform ORO notifications.

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

NEI 99-01 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.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the Commonwealth of Pennsylvania, Luzerne and Columbia County EOCs

The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

Susquehanna Basis Reference(s):

1. EP-RM-007 Emergency Telephone Instructions and Directory
2. SSES Emergency Plan Section 8
3. FSAR Section 9.5.2
4. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 8 – Hazardous Event Affecting Safety Systems

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

EAL:

SA8.1 **Alert**

The occurrence of **any** Table S-4 hazardous event

AND

EITHER:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Table S-4 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

Mode Applicability:

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

Definition(s):

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 require a post-event inspection to determine if the attributes of an explosion are present.

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

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

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 (as defined in 10CFR50.2):

> Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:
>
> (1) The integrity of the reactor coolant pressure boundary;
>
> (2) The capability to shut down the reactor and maintain it in a safe shutdown condition;
>
> (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

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.

Susquehanna Basis:

- The significance of a seismic event is discussed under EAL HU2.1 (ref. 1, 2).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 3, 4, 5).
- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 80 mph. (ref. 6, 7).
- Areas containing functions and systems required for safe shutdown of the plant are identified by Fire Zone in the fire response procedure (ref. 8, 9).
- An EXPLOSION that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

NEI 99-01 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.

The first condition 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.

The second condition 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.

Susquehanna Basis Reference(s):

1. ON-000-002 Severe Weather / Natural Phenomena
2. FSAR Section 3.7 Seismic Design
3. ON-169(269)-001 Flooding in Turbine Building
4. ON-169(269)-002 Flooding in Reactor Building
5. FSAR Section 3.4 Water Level (Flood) Design
6. FSAR Section 3.3 Wind and Tornado Loadings
7. FSAR Section 3.5 Missile Projection
8. SSES-FPRR Section 6.2 Fire Area Description
9. ON-013-001 Response to Fire
10. NEI 99-01 SA9

Category E – Independent Spent Fuel Storage Installation (ISFSI)

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

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

An Unusual Event is declared on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated.

A hostile security event that leads to a potential loss in the level of safety of the ISFSI is a classifiable event under Security category EAL HS1.1.

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

Category: E - ISFSI

Sub-category: None

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Notification of Unusual Event

Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by a radiation reading on a loaded spent fuel cask > **any** of the following:

- 800 mrem/hr at 3 ft from the HSM surface
- 200 mrem/hr on contact on the outside of the HSM door centerline
- 40 mrem/hr on contact on the end shield wall exterior

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY – The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the Susquehanna ISFSI, Confinement Boundary is defined as the Dry Shielded Canister (DSC).

Susquehanna Basis:

The SSES Independent Spent Fuel Storage Installation utilizes the standardized NUHMOS Horizontal Modular System. The standardized NUHMOS® system is a horizontal canister system composed of a steel dry shielded canister (DSC) and a reinforced concrete horizontal storage module (HSM). The DSC provides confinement and criticality control for the storage and transfer of irradiated fuel. The HSM houses the DSC and provides for heat removal. An HSM is considered loaded when it houses a DSC containing spent fuel. (ref. 1, 2)

The values shown represent 2 times the limits specified in the ISFSI Certificate of Compliance Technical Specification 1.2.7, HSM Dose Rates (ref. 1).

NEI 99-01 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 technical specification multiple of "2 times" 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.

Susquehanna Basis Reference(s):

1. Certification of Compliance No. 1004 for the NUHOMS® Storage System
2. TRM B3.10.3 Independent Spent Fuel Storage Installation (ISFSI)
3. NEI 99-01 E-HU1

Category F – Fission Product Barrier Degradation

EAL Group: Hot Conditions (RCS temperature > 200°F); EALs in this category are applicable only in one or more hot operating modes.

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

A. Fuel Clad (FC): The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.

B. Reactor Coolant System (RCS): The RCS Barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping up to and including the isolation valves.

C. Containment (PC): The drywell, the wetwell, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves comprise the PC barrier. Primary Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to either a Site Area Emergency or a General Emergency.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 2). "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Alert:
Any loss or any potential loss of either Fuel Clad or RCS

Site Area Emergency:
Loss or potential loss of any two barriers

General Emergency:
Loss of any two barriers and loss or potential loss of third barrier

The logic used for emergency classification based on fission product barrier monitoring should reflect the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Primary Containment Barrier.
- Unusual Event ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.
- For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to

ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC RG1 has been exceeded.

- The fission product barrier thresholds specified within a scheme reflect SSES design *and operating characteristics.*
- As used in this category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside the primary containment, an interfacing system, or outside of the primary containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered RCS leakage.
- At the Site Area Emergency level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the Emergency Director/Recovery Manager would have more assurance that there was no immediate need to escalate to a General Emergency.

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss of **any** two barriers and loss or potential loss of the third barrier

EAL:

FG1.1	**General Emergency**
Loss of **any** two barriers **AND** Loss or potential loss of the third barrier (Table F-1)	

Mode Applicability:

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

Definition(s):

None

Susquehanna Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the General Emergency classification level each barrier is weighted equally. A General Emergency is therefore appropriate for any combination of the following conditions:

- Loss of Fuel Clad, RCS and Containment barriers
- Loss of Fuel Clad and RCS barriers with potential loss of Containment barrier
- Loss of RCS and Containment barriers with potential loss of Fuel Clad barrier
- Loss of Fuel Clad and Containment barriers with potential loss of RCS barrier

NEI 99-01 Basis:

None

Susquehanna Basis Reference(s):

1. NEI 99-01 FG1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss or potential loss of **any** two barriers

EAL:

FS1.1	**Site Area Emergency**
Loss or potential loss of **any** two barriers (Table F-1)	

Mode Applicability:

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

Definition(s):

None

Susquehanna Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Site Area Emergency classification level, each barrier is weighted equally. A Site Area Emergency is therefore appropriate for any combination of the following conditions:

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

At the Site Area Emergency classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, the Emergency Director/Recovery Manager would have greater assurance that escalation to a General Emergency is less imminent.

NEI 99-01 Basis:

None

Susquehanna Basis Reference(s):

1. NEI 99-01 FS1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **Any** loss or **any** potential loss of either Fuel Clad or RCS

EAL:

FA1.1	**Alert**
Any loss or **any** potential loss of **EITHER** Fuel Clad or RCS (Table F-1)	

Mode Applicability:

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

Definition(s):

None

Susquehanna Basis:

Fuel Clad, RCS and Primary Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Alert classification level, Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1

NEI 99-01 Basis:

None

Susquehanna Basis Reference(s):

1. NEI 99-01 FA1

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Introduction

Table F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Containment). The table is structured so that each of the three barriers occupies adjacent columns. Each fission product barrier column is further divided into two columns; one for Loss thresholds and one for Potential Loss thresholds.

The first column of the table (to the left of the Fuel Clad Loss column) lists the categories (types) of fission product barrier thresholds. The fission product barrier categories are:

A. RPV Level

B. RCS Leak Rate

C. Primary Containment Conditions

D. Primary Containment Radiation / RCS Activity

E. Primary Containment Integrity or Bypass

F. ED/RM Jugement

Each category occupies a row in Table F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word "None" is entered in the cell.

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A would be assigned "FC Loss A.1," the third Containment barrier Potential Loss would be assigned "PC P-Loss C.3," etc.

If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of Table F-1, locates the likely category and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine if a threshold has been exceeded. If a threshold has not been exceeded, the EAL-user proceeds to the next likely category and continues review of the thresholds in the new category

If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded, only that one barrier is lost or potentially lost. The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if containment radiation is sufficiently high, a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Containment barrier can occur. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, and FA1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad barrier threshold bases appear first, followed by the RCS barrier and finally the Containment barrier threshold bases. In each barrier, the bases are given according category Loss followed by category Potential Loss beginning with Category A, then B,..., F.

Table F-2 provides a human factors enhancement mechanism to track the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Containment) to assist with quickly determining which initiating condition for EALs FG1.1, FS1.1, or FA1.1 is met.

<table>
<tr><th colspan="7">Table F-1 Fission Product Barrier Threshold Matrix</th></tr>
<tr><th></th><th colspan="2">Fuel Clad Barrier</th><th colspan="2">Reactor Coolant System Barrier</th><th colspan="2">Primary Containment Barrier</th></tr>
<tr><th>Category</th><th>Loss</th><th>Potential Loss</th><th>Loss</th><th>Potential Loss</th><th>Loss</th><th>Potential Loss</th></tr>
<tr><td>A
RPV Level</td><td>1. SAGs entered</td><td>1. RPV level CANNOT BE RESTORED AND MAINTAINED > -161 in. or CANNOT be determined</td><td>1. RPV level CANNOT BE RESTORED AND MAINTAINED > -161 in. or CANNOT be determined</td><td>None</td><td>None</td><td>1. SAGs enetered</td></tr>
<tr><td>B
RCS Leak Rate</td><td>None</td><td>None</td><td>1. UNISOLABLE break in any of the following:
• Main Steam Line
• HPCI Steam Line
• RCIC Steam Line
• RWCU
• Feedwater
OR
2. Emergency RPV Depressurization is required</td><td>1. UNISOLABLE primary system leakage that results in exceeding EITHER of the following:
• One or more Max Normal Reactor Building Radiation Limits (EO-000-104 Table 9) that can be read in the Control Room (Table F-3)
OR
• One or more Max Normal Reactor Building area temperature Limits (EO-000-104 Table 8) that can be read in the Control Room (Table F-4)</td><td>1. UNISOLABLE primary system leakage that results in exceeding EITHER of the following:
• One or more Max Safe Reactor Building Radiation Limits (EO-000-104 Table 9) that can be read in the Control Room (Table F-5)
OR
• One or more Max Safe Reactor Building area temperature Limits (EO-000-104 Table 8) that can be read in the Control Room (Table F-6)</td><td>None</td></tr>
<tr><td>C
PC Conditions</td><td>None</td><td>None</td><td>1. Primary Containment pressure > 1.72 psig due to RCS leakage</td><td>None</td><td>1. UNPLANNED rapid drop in Primary Containment pressure following Primary Containment pressure rise
OR
2. Primary Containment pressure response not consistent with LOCA conditions</td><td>1. Primary Containment pressure > 53 psig
OR
2. Deflagration concentrations exist inside PC ($H_2 \geq 6\%$ AND $O_2 \geq 5\%$)
OR
3. Heat Capacity Temperature Limit (HCTL) exceeded</td></tr>
<tr><td>D
PC Rad / RCS Activity</td><td>1. CHRRM radiation > 3.0E+3 R/hr
OR
2. Primary coolant activity > 300 μCi/gm I-131 dose equivalent</td><td>None</td><td>1. CHRRM radiation > 7.0E+0 R/hr with indications of RCS leakage inside the drywell</td><td>None</td><td>None</td><td>1. CHRRM radiation > 4.0E+4 R/hr</td></tr>
<tr><td>E
PC Integrity or Bypass</td><td>None</td><td>None</td><td>None</td><td>None</td><td>1. UNISOLABLE direct downstream pathway to the environment exists after Primary Containment isolation signal
OR
2. Intentional Primary Containment venting per EP-DS-004 RPV and PC Venting</td><td>None</td></tr>
<tr><td>F
ED/RM Judgment</td><td>1. Any condition in the opinion of the Emergency Director/Recovery Manager that indicates loss of the fuel clad barrier</td><td>1. Any condition in the opinion of the Emergency Director/Recovery Manager that indicates potential loss of the fuel clad barrier</td><td>1. Any condition in the opinion of the Emergency Director/Recovery Manager that indicates loss of the RCS barrier</td><td>1. Any condition in the opinion of the Emergency Director/Recovery Manager that indicates potential loss of the RCS barrier</td><td>1. Any condition in the opinion of the Emergency Director/Recovery Manager that indicates loss of the Primary Containment barrier</td><td>1. Any condition in the opinion of the Emergency Director/Recovery Manager that indicates potential loss of the Primary Containment barrier</td></tr>
</table>

Table F-2 Fission Product Barrier Status Table																				
Circle the X's in the table below for all applicable situations. Declare the EAL based upon all circled X's in any column.	**FG1: General Emergency** Loss of **any** two barriers and loss or potential loss of third barrier.				**FS1: Site Area Emergency** Loss or potential loss of **any** two barriers												FA1: Alert **Any** loss or **any** potential loss of **EITHER** Fuel Clad or RCS barrier			
Fuel Clad – Loss	X	X		X	X		X		X	X							X			
Fuel Clad – Potential Loss			X			X		X			X	X						X		
RCS – Loss	X	X	X		X			X					X	X					X	
RCS – Potential Loss				X		X	X								X	X				X
Primary Containment – Loss	X		X	X					X		X		X		X					
Primary Containment – Potential Loss		X								X		X		X		X				

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Loss

Threshold:

1. SAGs entered

Definition(s):

N/A

Susquehanna Basis:

EOPs specify the requirement for entry to the SAGs when core cooling is severely challenged. These EOPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined. SAG entry is required when (ref. 1, 2):

- RPV water level CANNOT BE RESTORED AND MAINTAINED above -179 in. (MSCRWL)

  AND

- RPV water level above -210 in. (jet pump suction) with at least one core spray loop injecting into the RPV at > 6350 gpm

OR

- TSC confirmation that core damage is progressing due to inadequate core cooling

The above EOP conditions represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

SAGs entry is also a Potential Loss of the Primary Containment barrier (PC P-Loss A.1) which constitutes a Site Area Emergency. The threshold for the RCS barrier (RCS Loss A.1) should also be evaluated for escalation to a General Emergency if SAGs entry results in meeting that threshold.

NEI 99-01 Basis:

The Loss threshold represents the EOP requirement for SAG entry. This is identified in the BWROG EPGs/SAGs when the phrase, "SAG Entry Is Required," appears. Since a site-specific RPV water level is not specified here, the Loss threshold phrase, "SAG entry required," also accommodates the EOP need to flood the primary containment when RPV water level cannot be determined and core damage due to inadequate core cooling is believed to be occurring.

Susquehanna Basis Reference(s):

1. EO-000-102 RPV Control
2. EO-000-114 RPV Flooding
3. NEI 99-01 RPV Water Level Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

1. RPV level **CANNOT** BE RESTORED AND MAINTAINED > -161 in. or **CANNOT** be determined

Definition(s):

N/A

Susquehanna Basis:

An RPV water level instrument reading of -161 in. indicates RPV level is at the top of active fuel (TAF) (ref. 1). When RPV level is at or above the TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV water level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops to TAF, the level is indicative of a challenge to core cooling and the Fuel Clad barrier.

This Fuel Clad barrier Potential Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

When RPV water level cannot be determined, EOPs require entry to EO-000-114, RPV Flooding. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained (ref. 2). When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EO-000-114 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in scram-failure events. If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the fuel clad barrier exists. This threshold is intended to be interpreted the same as in EO-000-114, that is, a loss of instrumentation is not, by itself, a loss of ability to determine level.

Note that EO-000-113, Level/Power Control, may require intentionally lowering RPV water level to -161 in. and control level between the Minimum Steam Cooling RPV Water Level (MSCRWL) and the top of active fuel (ref. 3). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs SA6 or SS6 will dictate the need for emergency classification.

NEI 99-01 Basis:

This water level corresponds to the top of the active fuel and is used in the EOPs to indicate a challenge to core cooling.

The RPV water level threshold is the same as RCS barrier Loss threshold A.1. Thus, this threshold indicates a Potential Loss of the Fuel Clad barrier and a Loss of the RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

This threshold is considered to be exceeded when, as specified in the site-specific EOPs, RPV water CANNOT BE RESTORED AND MAINTAINED above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this Fuel Clad barrier Potential Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

The term "CANNOT BE RESTORED AND MAINTAINED above" means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value *CANNOT BE RESTORED AND MAINTAINED* above a specified limit does not require immediate action simply because the current value is below the top of active fuel, but does not permit extended operation below the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs SA6 or SS6 will dictate the need for emergency classification.

Since the loss of ability to determine if adequate core cooling is being provided presents a significant challenge to the fuel clad barrier, a potential loss of the fuel clad barrier is specified.

Susquehanna Basis Reference(s):

1. EO-000-102 RPV Control
2. EO-000-114 RPV Flooding
3. EO-000-113 Level/Power Control
4. NEI 99-01 RPV Water Level Fuel Clad Potential Loss 2.A

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. CHRRM radiation > 3.0E+3 R/hr

Definition(s):

None

Susquehanna Basis:

Two gamma radiation levels ion chambers are located inside the drywell to monitor post-accident radiation levels. The detectors are located in the drywell on elevation 719'. They are always in service and read out on the C601 panel. Range is 10^0 to 10^8 R/hr. Environmentally qualified since they are required for use after an accident. A U-234 bug source is installed in the detector to serve as a self-check for instrument operability. The source provides approximately a 1 R/hr dose rate with the reactor shutdown. When the plant is at 100% power, drywell rad indication is normally about 3-4 R/hr. A reading of 3,000 R/hr indicates the release of reactor coolant into the drywell with elevated activity indicative of fuel clad damage (ref. 2).

For a fuel failure event equivalent to approximately 1% of cladding failure and an instantaneous and complete release of reactor coolant to the primary containment, the response of the Containment High Range Radiation Monitors (CHRRM) in the drywell will be approximately 3,450 R/hr immediately after shutdown (rounded to 3,000 R/hr, which approximates the dose rate 10 minutes after shutdown). This assumes that the release has occurred soon after reactor shutdown, and that the fuel cladding failures produce a coolant source term of 300 μCi/gm of I-131 dose equivalent just prior to the release into primary containment. (ref. 2)

NEI 99-01 Basis:

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals 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 1% 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 radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold D.1 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.

There is no Fuel Clad Barrier Potential Loss threshold associated with Primary Containment Radiation.

Susquehanna Basis Reference(s):

1. General Atomic High-Range Gamma Radiation Monitoring System Manual
2. Calculation EC RADN 0525 Rev 2, "Estimation of Containment High Range Radiation Monitor Response to a Loss of Coolant Accident for Emergency Planning Purposes," January 8, 2008
3. NEI 99-01 Primary Containment Radiation Fuel Clad Loss 4.A

Barrier: Fuel Clad

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

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

Susquehanna Basis:

The Fuel Clad Barrier shall be declared "lost" if the Primary coolant activity is determined to be > 300 µCi/gm I-131 dose equivalent. Two separate methods can be used make this determination:

- Sample collection and analysis of reactor coolant activity
- Analysis of plant parameters to determine fuel clad damage > 1%

Sample collection and analysis of reactor coolant activity are accomplished in accordance with CH-ON-007.

It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. The fuel clad damage analysis methodology in ref. 3 provides an alternative method to determine if the I-131 dose equivalent activity is > 300 µCi/gm.

Fuels Engineering determines if ≥ 1% fuel clad damage has occurred based on the analysis methodology in ref. 3. Fuel clad damage equal to 1% corresponds to at least 300 micro-Ci/gm I-131 dose equivalent in the reactor coolant, and drywell radiation values of at least 3000 R/hr during LOCA events (breach inside primary containment) (ref. 1). However, drywell radiation levels can be significantly lower than 3000 R/hr with a similar amount of fuel damage without a LOCA (no breach inside primary containment), since the fission products remain in the reactor vessel/do not escape into the drywell space and the CHRM is shielded from the radiation source (ref. 2 and 3).

NEI 99-01 Basis:

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 1% 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.

There is no Potential Loss threshold associated with RCS Activity.

Susquehanna Basis Reference(s):

1. Calculation EC RADN 0525 Rev 2, "Estimation of Containment High Range Radiation Monitor Response to a Loss of Coolant Accident for Emergency Planning Purposes," January 8, 2008
2. EWR 1642196 – Investigation of EPLAN 1% Clad Damage vs. Barrier Loss

3. EP-PS-324 – Fuels Lead Engineer/Core Thermal Hydraulics Engineer
4. CH-ON-007 – Emergency Sampling Procedures
5. NEI 99-01 RCS Activity Fuel Clad Loss 1.A

Barrier: Fuel Clad

Category: D. PC Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. PC Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: F. ED/RM Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Director/Recovery Manager that indicates loss of the Fuel Clad barrier

Definition(s):

None

Susquehanna Basis:

The Emergency Director/Recovery Manager judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director/Recovery Manager should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

NEI 99-01 Basis:

This threshold addresses any other factors that are to be used by the Emergency Director/Recovery Manager in determining whether the Fuel Clad barrier is lost

Susquehanna Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Fuel Clad Loss 6.A

Barrier: Fuel Clad

Category: F. ED/RM Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Director/Recovery Manager that indicates potential loss of the Fuel Clad barrier

Susquehanna Basis:

The Emergency Director/Recovery Manager judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director/Recovery Manager should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

NEI 99-01 Basis:

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

Susquehanna Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Potential Fuel Clad Loss 6.A

Barrier: Reactor Coolant System

Category: A. RPV Level

Degradation Threat: Loss

Threshold:

1. RPV level **CANNOT** BE RESTORED AND MAINTAINED > -161 in. or **CANNOT** be determined

Definition(s):

None

Susquehanna Basis:

An RPV level instrument reading of -161 in. indicates RPV level is at the top of active fuel (TAF) (ref. 1). TAF is significantly lower than the normal operating RPV level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Primary Containment barriers, and initiation of all ECCS. If RPV level cannot be maintained above TAF, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the decreasing level trend. The cause of the loss of RPV inventory is therefore assumed to be a LOCA. By definition, a LOCA event is a Loss of the RCS barrier.

This RCS barrier Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

When RPV water level cannot be determined, EOPs require entry to EO-000-114, RPV Flooding. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained (ref. 2). When all means of determining RPV water level are unavailable, the RCS barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EO-000-114 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in scram-failure events. If RPV water level cannot be determined with respect to the top of active fuel, a loss of the RCS barrier exists. This threshold is intended to be interpreted the same as in EO-000-114, that is, a loss of instrumentation is not, by itself, a loss of ability to determine level.

The conditions of this threshold are also a Potential Loss of the Fuel Clad barrier (FC P-Loss A.1). A Loss of the RCS barrier and Potential Loss of the Fuel Clad barrier requires a Site Area Emergency classification.

Note that EO-000-113, Level/Power Control, may require intentionally lowering RPV water level to -161 in. and control level between the Minimum Steam Cooling RPV Water Level (MSCRWL) and the top of active fuel (ref. 3). Although such action is a challenge to core cooling and the

Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs SA6 or SS6 will dictate the need for emergency classification.

NEI 99-01 Basis:

This water level corresponds to the top of active fuel and is used in the EOPs to indicate challenge to core cooling.

The RPV water level threshold is the same as Fuel Clad barrier Potential Loss threshold A.1. Thus, this threshold indicates a Loss of the RCS barrier and Potential Loss of the Fuel Clad barrier and that appropriately escalates the emergency classification level to a Site Area Emergency.

This threshold is considered to be exceeded when, as specified in the site-specific EOPs, RPV water CANNOT BE RESTORED AND MAINTAINED above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this RCS barrier Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

Susquehanna Basis Reference(s):

1. EO-000-102 RPV Control
2. EO-000-114 RPV Flooding
3. EO-000-113 Level/Power Control
4. NEI 99-01 RPV Water Level RCS Loss 2.A

Barrier: Reactor Coolant System

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

1. UNISOLABLE break in **ANY** of the following:
   - Main Steam Line
   - HPCI Steam Line
   - RCIC Steam Line
   - RWCU
   - Feedwater

Definition(s):

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

- The term UNISOLABLE also includes any decision by plant staff or procedure direction to not isolate a primary system.
- Normal leakage past a closed isolation valve is not considered UNISOLABLE leakage.

Susquehanna Basis:

As used in this threshold, local isolation actions can only be credited if isolation can be completed promptly (i.e. within 15 min.).

In the case of a failure of both isolation valves to close but in which no downstream flow path exists, emergency declaration under this threshold would not be required. Similarly, if the emergency response requires the normal process flow of a system outside primary containment (e.g., EOP requirement to bypass MSIV low RPV water level interlocks and maintain the main condenser as a heat sink using main turbine bypass valves), the threshold is not met. The combination of these threshold conditions represent the loss of both the RCS and Primary Containment (see PC Loss E.1) barriers and justifies declaration of a Site Area Emergency (i.e., Loss or Potential Loss of any two barriers).

High steam flow and high steam tunnel temperature annunciators are an indication of a Main Steam Line break. Either parameter will cause an isolation of the MSIVs. Note that the high steam flow alarm may clear if any of the MSIVs close and flow is reduced below the setpoint. If the high steam flow alarm was received (even though it was subsequently cleared) or there is other indication of high flow along with the high temperature alarm, the entry condition for this threshold has been met.

Even though RWCU and Feedwater systems do not contain steam, they are included in the list because an UNISOLABLE break could result in the high-pressure discharge of fluid that is flashed to steam from relatively large volume systems directly connected to the RCS. Note: Each of the two feedwater injection lines is isolated from the reactor vessel via a series of swing check valves. The ability of these check valves to isolate cannot be determined until after feedwater is no longer injecting into the reactor vessel.

NEI 99-01 Basis:

Large high-energy lines that rupture outside primary containment can discharge significant amounts of inventory and jeopardize the pressure-retaining capability of the RCS until they are isolated. If it is determined that the ruptured line cannot be promptly isolated, the RCS barrier Loss threshold is met.

Susquehanna Basis Reference(s):

1. FSAR Section 5.4.5
2. FSAR Section 6.3
3. FSAR Section 5.4.6
4. FSAR Section 10.4.7
5. FSAR Section 5.4.8
6. P&ID M-141 Nuclear Boiler
7. NEI 99-01 RCS Leak Rate RCS Loss 3.A

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

2. Emergency RPV Depressurization is required

Definition(s):

N/A

Susquehanna Basis:

Plant symptoms requiring Emergency RPV Depressurization per the EOPs are indicative of a loss of the RCS barrier. If emergency depressurization is required, the plant operators are directed to open safety relief valves (SRVs). Even though the RCS is being vented into the suppression pool, a loss of the RCS exists due to the diminished effectiveness of the RCS pressure barrier to a release of fission products beyond its boundary.

NEI 99-01 Basis:

Emergency RPV Depressurization in accordance with the EOPs is indicative of a loss of the RCS barrier. If Emergency RPV Depressurization is performed, the plant operators are directed to open safety relief valves (SRVs) and keep them open. Even though the RCS is being vented into the suppression pool, a Loss of the RCS barrier exists due to the diminished effectiveness of the RCS to retain fission products within its boundary.

Susquehanna Basis Reference(s):

1. EO-000-102 RPV Control
2. EO-000-103 Primary Containment Control
3. EO-000-104 Secondary Containment Control
4. EO-000-105 Radioactivity Release Control
5. EO-000-112 Emergency RPV Depressurization
6. EO-000-113 Level Power Control
7. EO-000-114 RPV Flooding
8. NEI 99-01 RCS Leak Rate RCS Loss 3.B

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

1. UNISOLABLE primary system leakage that results in exceeding **EITHER** of the following:
   - One or more Max Normal Reactor Building Radiation Limits (EO-000-104 Table 9) that can be read in the control room (Table F-3)

     OR
   - One or more Max Normal Reactor Building Area Temperature Limits (EO-000-104 Table 8) that can be read in the control room (Table F-4)

Table F-3

Max Normal Reactor Building Radiation Limits

RB Area Elevation (ft)	ARM Number	ARM Channel Description	Max Norm Rad Limit
818	35 14 15 42 47 (44 U2)	Cask Stor Area Spent Fuel Crit Mon Refuel Floor North (South U2) Refuel Floor West Spent Fuel Crit Mon	Hi Alarm
749	8 10 11	RWCU Recirc PP Access Fuel Pool PP Area Rx Bldg Sample St	Hi Alarm
719	5 6	CRD North CRD South	Hi Alarm
670	16	Remote Shutdown Room	Hi Alarm
645	3 2 25 1 4	HPCI PP & Turbine Room RCIC PP & Turbine Room RHR A C PP Room RHR B D PP Room RB/RW Sump Area	Hi Alarm

Table F-4

Max Normal Reactor Building Temperature Limits

RB Area Elevation (ft)	Area Temperature	Max Normal Temp (°F)
749	RWCU-Pump Room RWCU-Heat Exch Room RWCU-Penetration Room	120 120 120
719	Main Steam Line Tunnel	157
683	HPCI Pipe Routing Area RCIC Pipe Routing Area	120 120
645	HPCI-Equip Area HPCI-Emerg Area Cooler	120 120
645	RCIC-Emerg Area Cooler RCIC-Equip Area	120 120
645	RHR Equip Area 1	110
645	RHR Equip Area 2	110

Definition(s):

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

- The term UNISOLABLE also includes any decision by plant staff or procedure direction to not isolate a primary system *(e.g. EP-000-104 has direction not to isolate systems if they are needed for EOP actions or damage control)*
- Normal leakage past a closed isolation valve is not considered UNISOLABLE leakage.

Susquehanna Basis:

This RCS Potential Loss threshold is limited to primary system leakage that results in exceeding one or more Max Normal Reactor Building Radiation or Temperature Limits that can be remotely determined from within the control room (ref. 1).

NRC regulations require the SSES 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.

Max Normal conditions shall be assumed to be from RCS leakage until proven otherwise.

- The 15 minute classification requirement (EAL "trigger") for this threshold begins when one or more of the above Max Normal Reactor Building Radiation or Temperature Limits are exceeded.

- o If subsequent actions taken to isolate the leak are successful within the 15 minute classification period, this EAL should not be declared. (Note: EAL SU5.1 should be evaluated).
- o If subsequent investigation, within the 15 minute classification period, reveals that the Max Norm conditions are not due to RCS leakage, this EAL should not be declared.

Note that a RCS leak could cause the fire suppression systems to actuate. If this occurs, the potential exists that the fire suppression systems could cause the area temperatures to be lower than the values specified in the EO-000-104 even though there is a RCS leak.

- Once it is known that the cause of exceeding Max Norm temperatures is due to a FIRE or loss of ventilation then this threshold is not met. The applicable FIRE EAL should be evaluated.
- If there is certainty that the fire suppression system actuation was caused by a RCS leak and NOT a fire then it is appropriate to judge that the MAX NORM temperature limits have been met even if the actual area temperatures are lower than those listed in the EO-000-104.

The presence of elevated general area temperatures or radiation levels in the secondary containment may be indicative of UNISOLABLE primary system leakage outside the primary containment.

The Max Normal Reactor Building Limit values define this RCS threshold because they are the maximum normal operating values and signify the onset of abnormal system operation. When parameters reach this level, equipment failure or misoperation may be occurring. Elevated parameters may also adversely affect the ability to gain access to or operate equipment within the affected area. The locations into which the primary system discharge is of concern correspond to the areas addressed in EO-000-104, Secondary Containment Control, Table 8 that can be read in the control room (ref. 2).

Cycling of SRVs to reduce primary system overpressure is not considered reactor coolant leakage. Inventory loss events, such as a stuck open SRV, venting and draining the RCS during cold shutdown or refueling, should not be considered when referring to "RCS leakage" because they are not indications of a break which could propagate. For these events entry into this threshold is not warranted however consideration should be given to RCS Loss – RCS Leak Rate threshold B2.

In general, multiple indications should be used to validate that a primary system is discharging outside Primary Containment. For example, a high area radiation condition may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room flooding, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the secondary containment.

NEI 99-01 Basis:

Potential loss of RCS based on primary system leakage outside the primary containment is determined from EOP temperature or radiation Max Normal Operating values in areas such as main steam line tunnel, RCIC, HPCI, etc., which indicate a direct path from the RCS to areas outside primary containment.

A Max Normal Operating value is the highest value of the identified parameter expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

The indicators reaching the threshold barriers and confirmed to be caused by RCS leakage from a primary system warrant an Alert classification. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

An UNISOLABLE leak as described above escalates to a Site Area Emergency when combined with Containment Barrier Loss threshold B.1 (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

Susquehanna Basis Reference(s):

1. NCV 05000387; 388/2013005-04, Inadequate Instrumentation to Implement EALs for Fission Product Barrier Degradation
2. EO-000-104 Secondary Containment Control
3. NEI 99-01 RCS Leak Rate RCS Potential Loss 3.A

Barrier: Reactor Coolant System

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

1. Primary Containment pressure > 1.72 psig due to RCS leakage

Definition(s):

None

Susquehanna Basis:

The drywell high pressure scram setpoint is an entry condition to EO-000-102, RPV Control, and EO-000-103, Primary Containment Control (ref. 1, 2). Normal primary containment pressure control functions (e.g., operation of drywell coolers, vent through SGT, etc.) are specified in EO-000-103 in advance of less desirable but more effective functions (e.g., operation of drywell or suppression pool sprays, etc.).

In the design basis, primary containment pressures above the drywell high pressure scram setpoint are assumed to be the result of a high-energy release into the containment for which normal pressure control systems are inadequate or incapable of reversing the increasing pressure trend. Pressures of this magnitude, however, can be caused by non-LOCA events such as a loss of drywell cooling or inability to control primary containment vent/purge (ref. 3).

The threshold phrase "...due to RCS leakage" focuses the barrier failure on the RCS instead of the non-LOCA malfunctions that may adversely affect primary containment pressure. PC pressure greater than 1.72 psig with corollary indications (e.g., drywell temperature, indications of loss of RCS inventory) should therefore be considered a Loss of the RCS barrier. Loss of drywell cooling that results in pressure greater than 1.72 psig should not be considered an RCS barrier Loss.

NEI 99-01 Basis:

1.72 psig is the drywell high pressure setpoint which indicates a LOCA by automatically initiating the ECCS or equivalent makeup system.

There is no Potential Loss threshold associated with Primary Containment Pressure.

Susquehanna Basis Reference(s):

1. EO-000-102 RPV Control
2. EO-000-103 Primary Containment Control
3. FSAR Section 6.2.1 Primary Containment Functional Design
4. NEI 99-01 Primary Containment Pressure RCS Loss 1.A

Barrier: Reactor Coolant System

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. CHRRM radiation > 7.0E+0 R/hr with indication of a RCS leak inside the drywell

Definition(s):

N/A

Susquehanna Basis:

Two gamma radiation levels ion chambers are located inside the drywell to monitor post-accident rad levels. The detectors are located in the drywell on elevation 719'. They are always in service and read out on the C601 panel. Range is 10^0 to 10^8 R/hr. Environmentally qualified since they are required for use after an accident. A U-234 bug source is installed in the detector to serve as a self-check for instrument operability. The source provides approximately a 1 R/hr dose rate with the reactor shutdown (ref. 1). When the plant is at 100% power, drywell rad indication is normally about 3-4 R/hr. Containment High Range Radiation Monitor (CHRRM) readings of approximately 3 R/hr indicate an instantaneous release of reactor coolant at normal operating concentrations of I-131 to the drywell atmosphere. Adding this value to the normal CHRRM background readings of 3-4 R/hr (100% power normal operation) provides the value of 7 R/hr. (ref. 2)

Indication of a RCS leak into the drywell is added to qualify the radiation monitor indication to avoid declaring the loss of the RCS barrier for situations where the radiation increase is not due to a primary system leak. For situations that involve failure of the Fuel Clad Barrier alone, containment Radiation levels would increase to greater than 30 R/hr potentially giving a false indication of a loss of the RCS barrier. Therefore the EAL contains a qualifier to preclude over classification of the event if only the fuel clad barrier has failed. Indication of a leak should be determined by observing other containment indications such as sump level, drywell pressure and ambient temperature.

NEI 99-01 Basis:

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold D.1 since it indicates a loss of the RCS Barrier only.

There is no RCS Barrier Potential Loss threshold associated with Primary Containment Radiation.

Susquehanna Basis Reference(s):

1. General Atomic High-Range Gamma Radiation Monitoring System Manual

2. Calculation EC RADN 0525 Rev 2, "Estimation of Containment High Range Radiation Monitor Response to a Loss of Coolant Accident for Emergency Planning Purposes," January 8, 2008
3. NEI 99-01 Primary Containment Radiation RCS Loss 4.A

Barrier: Reactor Coolant System

Category: D. PC Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. PC Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: F. ED/RM Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Director/Recovery Manager that indicates loss of the RCS barrier

Definition(s):

None

Susquehanna Basis:

The Emergency Director/Recovery Manager judgment threshold addresses any other factors relevant to determining if the RCS barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to the recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director/Recovery Manager should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

NEI 99-01 Basis:

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

Susquehanna Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Loss 6.A

Barrier: Reactor Coolant System

Category: F. ED/RM Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Director/Recovery Manager that indicates potential loss of the RCS barrier

Definition(s):

None

Susquehanna Basis:

The Emergency Director/Recovery Manager judgment threshold addresses any other factors relevant to determining if the RCS barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to the inability to reach final safety acceptance criteria before completing all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director/Recovery Manager should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

NEI 99-01 Basis:

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

Susquehanna Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Potential Loss 6.A

Barrier: Primary Containment

Category: A. RPV Level

Degradation Threat: Loss

Threshold:

None

Barrier: Primary Containment

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

1. SAGs entered

Definition(s):

None

Susquehanna Basis:

EOPs specify the requirement for entry to the SAGs when core cooling is severely challenged. These EOPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined. SAG entry is required when (ref. 1, 2):

- RPV water level CANNOT BE RESTORED AND MAINTAINED above -179 in. (MSCRWL)

  AND

- RPV water level CANNOT BE RESTORED AND MAINTAINED above -210 in. (jet pump suction) with at least one core spray loop injecting into the RPV at > 6350 gpm

OR

- TSC confirmation that core damage is progressing due to inadequate core cooling

The above EOP conditions represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

SAGs entry is also a Potential Loss of the Fuel Clad barrier (FC P-Loss A.1) which constitutes a Site Area Emergency. The threshold for the RCS barrier (RCS Loss A.1) should also be evaluated for escalation to a General Emergency if SAGs entry results in meeting that threshold.

NEI 99-01 Basis:

The Potential Loss threshold is identical to the Fuel Clad Loss RPV Water Level threshold A.1. The Potential Loss requirement for SAG entry is required indicates adequate core cooling CANNOT BE RESTORED AND MAINTAINED and that core damage is possible. BWR EPGs/SAGs specify the conditions that require SAG entry. When SAG entry is required, the EPGs are exited and SAGs are entered. Entry into SAGs is a logical escalation in response to the inability to restore and maintain adequate core cooling.

PRA studies indicate that the condition of this Potential Loss threshold could be a core melt sequence which, if not corrected, could lead to RPV failure and increased potential for primary containment failure. In conjunction with the RPV water level Loss thresholds in the Fuel Clad and RCS barrier columns, this threshold results in the declaration of a General Emergency.

Susquehanna Basis Reference(s):

1. EO-000-102 RPV Control
2. EO-000-114 RPV Flooding
3. NEI 99-01 RPV Water Level PC Potential Loss 2.A

Barrier: Primary Containment

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

1. UNISOLABLE primary system leakage that results in exceeding **EITHER** of the following:
   - One or more Max Safe Reactor Building Radiation Limits (EO-000-104 Table 9) that can be read in the control room (Table F-5)

   OR

   - One or more Max Safe Reactor Building area temperature Limits (EO-000-104 Table 8) that can be read in the control room (Table F-6)

Table F-5

Max Safe Reactor Building Radiation Limits

RB Area Elevation (ft)	ARM Number	ARM Channel Description	Max Safe Rad Limit (R/HR)
818	49	Refuel Floor Area	10
749	52 54	RWCU Recirc PP Access Fuel Pool PP Area	10
719	50 51	CRD North CRD South	10
670	53	Remote Shutdown Room	10
645	48 57 55 56	HPCI PP & Turbine Room RCIC PP & Turbine Room RHR A C PP Room RHR B D PP Room	10

Table F-6

Max Safe Reactor Building Temperature Limits

RB Area Elevation (ft)	Area Temperature	Max Safe Temp (°F)
749	RWCU-Pump Room RWCU-Heat Exch Room RWCU-Penetration Room	147 147 131
719	Main Steam Line Tunnel	177
683	HPCI Pipe Routing Area RCIC Pipe Routing Area	167 167
645	HPCI-Equip Area HPCI-Emerg Area Cooler	167 167
645	RCIC-Emerg Area Cooler RCIC-Equip Area	167 167
645	RHR Equip Area 1	142
645	RHR Equip Area 2	142

Definition(s):

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

- The term UNISOLABLE also includes any decision by plant staff or procedure direction to not isolate a primary system *(e.g. EP-000-104 has direction not to isolate systems if they are needed for EOP actions or damage control)*
- Normal leakage past a closed isolation valve is not considered UNISOLABLE leakage.

Susquehanna Basis:

This Primary Containment Loss threshold is limited to primary system leakage that results in exceeding one or more Max Safe Reactor Building Radiation or Temperature Limits that can be remotely determined from within the control room (ref. 1).

NRC regulations require the SSES 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.

The 15 minute classification requirement (EAL "trigger") for this threshold begins when one or more of the above Max Safe Reactor Building Radiation or Temperature Limits are exceeded.

Max Safe conditions shall be assumed to be from RCS leakage until proven otherwise.

- If subsequent actions taken to isolate the leak are successful, **the threshold for this EAL is still met and must be declared**. Once leakage is isolated, downgrading the emergency may be appropriate.
- If subsequent investigation, within the 15 minute classification period, reveals that the Max Safe conditions are not due to RCS leakage, this EAL should not be declared.

Note that a RCS leak could cause the fire suppression systems to actuate. If this occurs the potential exists that the fire suppression systems could cause the area temperatures to be lower than the values specified in the EO-000-104 even though there is a RCS.

- Once it is known that the cause of exceeding MAX SAFE temperatures is due to a FIRE or loss of ventilation then this threshold is not met. The applicable FIRE EAL should be evaluated.
- If there is certainty that the fire suppression system actuation was caused by a RCS leak and NOT a fire then it is appropriate to judge that the MAX NORM temperature limits have been met even if the actual area temperatures are lower than those listed in the EO-000-104

The presence of elevated general area temperatures or radiation levels in the secondary containment may be indicative of UNISOLABLE primary system leakage outside the primary containment.

- The Maximum Safe Reactor Building Limit values define this Containment barrier threshold because they are indicative of problems in the secondary containment that are spreading and pose a threat to achieving a safe plant shutdown. This threshold addresses problematic discharges outside primary containment that may not originate from a high-energy line break. The locations into which the primary system discharge is of concern correspond to the areas addressed in EO-000-104, Secondary Containment Control, Table 8 that can be read in the control room (ref. 2).

In general, multiple indications should be used to validate that a primary system is discharging outside Primary Containment. For example, a high area radiation condition may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room flooding, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the secondary containment.

NEI 99-01 Basis:

The Max Safe Operating Temperature and the Max Safe Operating Radiation Level are each the highest value of these parameters at which neither: (1) equipment necessary for the safe shutdown of the plant will fail, nor (2) personnel access necessary for the safe shutdown of the plant will be precluded. EOPs utilize these temperatures and radiation levels to establish conditions under which RPV depressurization is required.

The temperatures and radiation levels should be confirmed to be caused by RCS leakage from a primary system. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

In combination with RCS Potential Loss B.1 this threshold would result in a Site Area Emergency.

There is no Potential Loss threshold associated with Primary Containment Isolation Failure.

Susquehanna Basis Reference(s):

1. NCV 05000387; 388/2013005-04, Inadequate Instrumentation to Implement EALs for Fission Product Barrier Degradation
2. EO-000-104 Secondary Containment Control
3. NEI 99-01 RCS Leak Rate PC Loss 3.C

Barrier: Primary Containment

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Primary Containment

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

1. UNPLANNED rapid drop in Primary Containment pressure following Primary Containment pressure rise

Definition(s):

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.

Susquehanna Basis:

None

NEI 99-01 Basis:

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

This threshold relies on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

Susquehanna Basis Reference(s):

1. NEI 99-01 Primary Containment Conditions PC Loss 1.A

Barrier: Primary Containment

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

2. Primary Containment pressure response **not** consistent with LOCA conditions

Definition(s):

None

Susquehanna Basis:

The calculated pressure response of the containment is shown in Figure 6.2-2 (ref. 1) (reproduced on next page). The maximum calculated drywell pressure (63.3 psia or 48.6 psig) is well below the design allowable pressure of 53 psig (ref. 2).

Due to conservatisms in LOCA analyses, actual pressure response is expected to be less than the analyzed response. For example, blowdown mass flowrate may be only 60-80% of the analyzed rate.

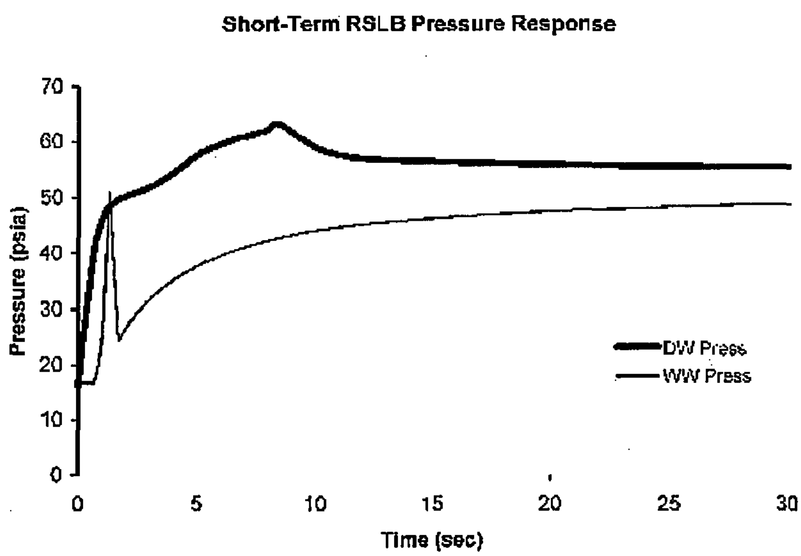
NEI 99-01 Basis:

Primary containment pressure should increase as a result of mass and energy release into the primary containment from a LOCA. Thus, primary containment pressure not increasing under these conditions indicates a loss of primary containment integrity.

This threshold relies on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

Susquehanna Basis Reference(s):

1. FSAR Figure 6.2-2
2. FSAR Section 6.2.1.1.3.1
3. NEI 99-01 Primary Containment Conditions PC Loss 1.B



FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
PRESSURE RESPONSE FOR RECIRCULATION LINE BREAK FIGURE 6.2-2, Rev. 56

Auto Cad: Figure Fsar 6_2_2.dwg

Barrier: Containment

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

1. Primary Containment pressure > 53 psig

Definition(s):

None

Susquehanna Basis:

When the primary containment exceeds the maximum allowable value (53 psig) (ref. 1), primary containment venting may be required even if offsite radioactivity release rate limits will be exceeded (ref. 2). The drywell and suppression chamber maximum allowable value of 53 psig is based on the primary containment design pressure as identified in the SSES accident analysis. If this threshold is exceeded, a challenge to the containment structure has occurred because assumptions used in the accident analysis are no longer valid and an unanalyzed condition exists. This constitutes a Potential Loss of the Containment barrier even if a containment breach has not occurred.

NEI 99-01 Basis:

The threshold pressure is the primary containment internal design pressure. Structural acceptance testing demonstrates the capability of the primary containment to resist pressures greater than the internal design pressure. A pressure of this magnitude is greater than those expected to result from any design basis accident and, thus, represent a Potential Loss of the Containment barrier.

Susquehanna Basis Reference(s):

1. FSAR Section 6.2.1.1.3.1
2. EO-000-103 Primary Containment Control
3. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.A

Barrier: Containment

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

2. Deflagration concentrations exist inside PC ($H_2 \geq 6\%$ **AND** $O_2 \geq 5\%$)

Definition(s):

None

Susquehanna Basis:

Deflagration (explosive) mixtures in the primary containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EPGs/SAGs indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to primary containment integrity. Hydrogen concentration of approximately 6% is considered the global deflagration concentration limit (ref. 1).

Except for brief periods during plant startup and shutdown, oxygen concentration in the primary containment is maintained at insignificant levels by nitrogen inertion. The specified values for this Potential Loss threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen) and readily recognizable because 6% hydrogen is well above the EO-000-103, Primary Containment Control, entry condition (ref. 2, 3). Values above the hydrogen/oxygen concentrations (6% and 5%, respectively) require intentional primary containment venting, which is defined to be a Loss of Containment (PC Loss E.2).

Hydrogen and oxygen concentration in the drywell during normal operation. These monitors are isolated by accident isolation signals. However, monitors CAC-AT-4409 and 4410 will be realigned to the primary containment for post-accident monitoring via an operator actuated isolation signal override circuit when directed by the EOPs.

NEI 99-01 Basis:

If hydrogen concentration reaches or exceeds the lower flammability limit, as defined in plant EOPs, in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside the primary containment, loss of the Containment barrier could occur.

Susquehanna Basis Reference(s):

1. BWROG EPG/SAG Revision 3, Subsection PC/G
2. EO-000-103 Primary Containment Control
3. EP-DS-001 Containment Combustible Gas Control

4. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.B

Barrier: Containment

Category: C. PC Conditions

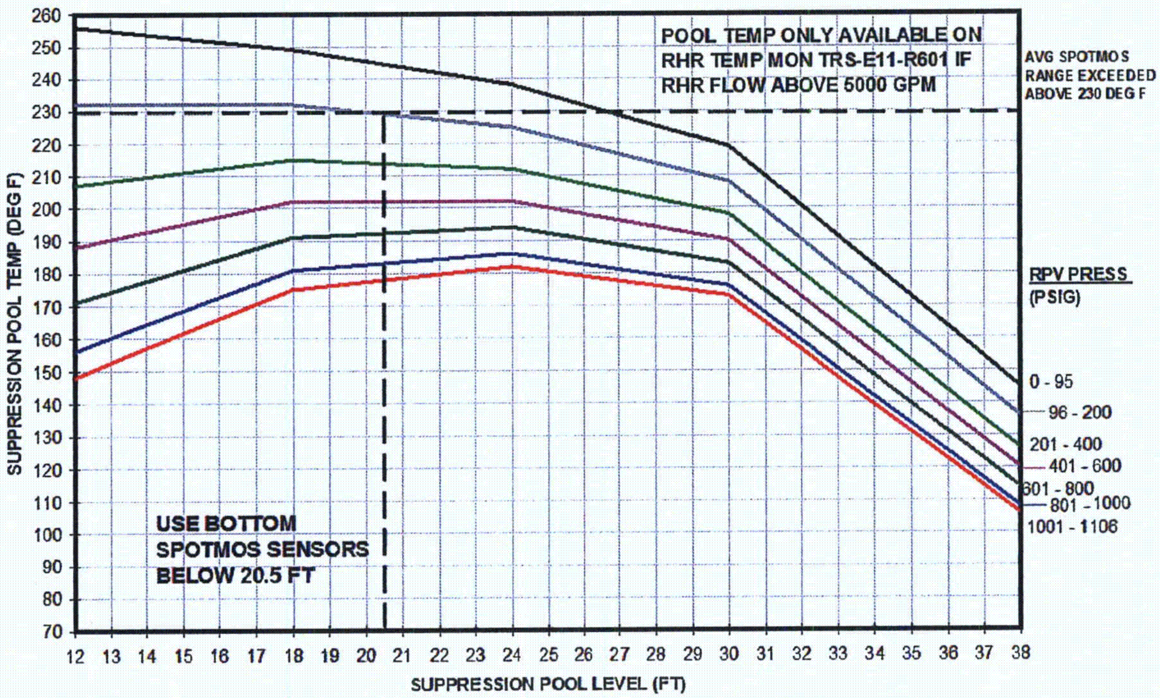
Degradation Threat: Potential Loss

Threshold:

3. Heat Capacity Temperature Limit (Figure - HCTL) exceeded

Figure – HCTL

Heat Capacity Temperature Limit



Definition(s):

None

Susquehanna Basis:

This threshold is met when the final step of section SP/T in EO-000-103, Primary Containment Control, is reached (ref. 1).

NEI 99-01 Basis:

The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool temperature from which Emergency RPV Depressurization will not raise:

> Suppression chamber temperature above the maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized,
>
> OR
>
> Suppression chamber pressure above Primary Containment Pressure Limit A, while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

The HCTL is a function of RPV pressure, suppression pool temperature and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant and therefore, the inability to maintain plant parameters below the limit constitutes a potential loss of containment.

Susquehanna Basis Reference(s):

1. EO-000-103 Primary Containment Control
2. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.C

Barrier: Primary Containment

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Primary Containment

Category: D. PC Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

1. CHRRM radiation > 4.0E+4R/hr

Definition(s):

None

Susquehanna Basis:

Two gamma radiation levels ion chambers are located inside the drywell to monitor post-accident radiation levels. The detectors are located in the drywell on elevation 719'. They are always in service and read out on the C601 panel. Range is 10^0 to 10^8 R/hr. Environmentally qualified since they are required for use after an accident. A U-234 bug source is installed in the detector to serve as a self-check for instrument operability. The source provides approximately a 1 R/hr dose rate with the reactor shutdown (ref. 1). When the plant is at 100% power, Containment High Range Radiation Monitor (CHRRM) indication is normally about 3-4 R/hr. A reading of 40,000 R/hr indicates the release of reactor coolant into the drywell with elevated activity indicative of 20% fuel clad damage (ref. 2).

An analysis of the Containment High Range Radiation Monitor (CHRRM) response to a Loss-Of-Coolant Accident (LOCA) is given in reference 2. Results are summarized in Reference 2, Section 2 for the case of 1% clad damage. For this threshold a reactor shutdown time of 1 hour is conservatively assumed. Although CHRRM data is always available for emergency planning, a one hour shutdown time is conservative for times less than one hour and is applicable to the timing of the sequence of events for a LOCA that would result in both the release of reactor coolant activity to containment and damage to 20% of the fuel clad. From reference 2, Section 2, the calculated CHRRM dose rate at one hour after shutdown for a complete loss of reactor coolant activity to containment with 1% clad damage is 2160 R/hr. Multiplying this value by a factor of 20 to account for 20% clad damage results in a CHRRM dose rate of 43,200 R/hr. This value is rounded to 40,000 R/hr for human factors considerations.

A Containment High Range Rad Monitor reading > 40,000 R/hr is a value which indicates significant fuel damage well in excess of that required for loss of RCS and Fuel Clad. A major failure of fuel cladding which allows radioactive material to be released from the core into the reactor coolant could result in a major release of radioactivity requiring offsite protective actions. **Regardless of whether containment is challenged, this amount of activity in containment corresponding to a CHHRM reading > 40,000 R/hr, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted.**

In order to reach this Containment barrier Potential Loss threshold, a loss of the RCS barrier (RCS Loss D.1) and a loss of the Fuel Clad barrier (FC Loss D.1) have already occurred. **This threshold, therefore, represents at a General Emergency classification.**

NEI 99-01 Basis:

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

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.

Susquehanna Basis Reference(s):

1. General Atomic High-Range Gamma Radiation Monitoring System Manual
2. Calculation EC RADN 0525 Rev 2, "Estimation of Containment High Range Radiation Monitor Response to a Loss of Coolant Accident for Emergency Planning Purposes," January 8, 2008
3. NEI 99-01 Primary Containment Radiation Fuel Clad Potential Loss 1.D

Barrier: Primary Containment

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. UNISOLABLE direct downstream pathway to the environment exists after Primary Containment isolation signal

Definition(s):

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

- The term UNISOLABLE also includes any decision by plant staff or procedure direction to not isolate a primary system *(e.g. EP-000-104 has direction not to isolate systems if they are needed for EOP actions or damage control)*
- Normal leakage past a closed isolation valve is not considered UNISOLABLE leakage.

Susquehanna Basis:

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway directly to the environment.

The concern is the unisolable open pathway to the environment. A failure of the ability to isolate any one line with a pathway directly to the environment indicates a breach of primary containment integrity. Examples include:

- This EAL is applicable if plant operators attempt to close isolation valves before any automatic setpoints are reached, and both valves fail to close AND a downstream pathway to the environment exists. If subsequent actions are successful, downgrading the emergency may be appropriate.
- Main Steam Line breaks with two MSIVs in one line failing to close.
- HPCI, RCIC or RWCU steam line breaks with all isolation valves in one line failing to close.
- UNISOLABLE containment atmosphere vent paths.
- Automatic closure of both isolation valves in one line are disabled AND an isolation signal occurs AND a downstream pathway exits.

The adjective "direct" modifies "release pathway" to discriminate against release paths through interfacing liquid systems. The following examples <u>do not</u> meet the EAL threshold:

- If the main condenser is available with an UNISOLABLE main intact steam line, there may be releases through the steam jet air ejectors and gland seal exhausters. These pathways are monitored, however, and do not meet the intent of an UNISOLABLE direct release path to the environment. These minor releases are assessed using the Category R, Abnormal Rad Release / Rad Effluent, EALs.
- Leakage into a closed system (e.g. RHR, Core Spray) is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment.
- Normal leakage past a closed isolation valve is not considered UNISOLABLE leakage.

- SCRAM Discharge Volume valves fail to close. The SDV drains to the Reactor Building Sump, which creates the pathway to the environment.
- The existence of an in–line charcoal filter (SGTS) does not make a release path indirect since the filter is not effective at removing fission noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.
- EO-000-103, Primary Containment Control, Section PC/P may specify primary containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1). Under these conditions with a valid containment isolation signal, the Containment barrier should be considered lost.

Declaration of this EAL threshold constitutes a radiological release in progress.

NEI 99-01 Basis:

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems or minor release pathways, such as instrument lines, not protected by the Primary Containment Isolation System (PCIS).

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.

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

Susquehanna Basis Reference(s):

1. EO-000-103 Primary Containment Control
2. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.A

Barrier: Primary Containment

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

2. Intentional Primary Containment venting per EP-DS-004 RPV and PC Venting

Definition(s):

None

Susquehanna Basis:

EO-000-103, Primary Containment Control, may specify primary containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1). The threshold is met when the operator begins venting the primary containment in accordance with EP-DS-004, RPV and PC Venting, not when actions are taken to bypass interlocks prior to opening the vent valves (ref. 2).

Because the containment vent valves are not qualified for opening/reclosure in a post-accident environment, there is no guarantee that venting, once initiated, can be terminated. Thus it is assumed that once the vents are opened with a source term, they are not re-closed (ref. 3).

NEI 99-01 Basis:

EOPs may direct primary containment isolation valve logic(s) to be intentionally bypassed, even if offsite radioactivity release rate limits will be exceeded. Under these conditions with a valid primary containment isolation signal, the containment should also be considered lost if primary containment venting is actually performed.

Intentional venting of primary containment for primary containment pressure control to the secondary containment and/or the environment is a Loss of the Containment. Venting for primary containment pressure control when not in an accident situation (e.g., to control pressure below the drywell high pressure scram setpoint) does not meet the threshold condition.

Susquehanna Basis Reference(s):

1. EO-000-103 Primary Containment Control
2. EP-DS-004 RPV and PC Venting
3. NL-98-036, SSES Safety Evaluation for EP-DS-004, Primary Containment and RPV Venting
4. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.B

Barrier: Primary Containment

Category: E. PC Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Primary Containment

Category: F. ED/RM Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Director/Recovery Manager that indicates loss of the Primary Containment barrier

Definition(s):

None

Susquehanna Basis:

The Emergency Director/Recovery Manager judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director/Recovery Manager should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

NEI 99-01 Basis:

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

Susquehanna Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

Barrier: Primary Containment

Category: F. ED/RM Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Director/Recovery Manager that indicates potential loss of the Primary Containment barrier

Definition(s):

None

Susquehanna Basis:

The Emergency Director/Recovery Manager judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director/Recovery Manager should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

NEI 99-01 Basis:

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

Susquehanna Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

ATTACHMENT 3

Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases

Background

NEI 99-01 Revision 6 ICs AA3 (SSES RA3.2) and HA5 (SSES HA5.1) prescribe declaration of an Alert based on impeded access to rooms or areas (due to either area radiation levels or hazardous gas concentrations) where equipment necessary for normal plant operations, cooldown or shutdown is located. These areas are intended to be plant operating mode dependent. Specifically the Developers Notes For AA3 and HA5 states:

> *The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.*
>
> *The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).*

Further, as specified in IC HA5:

> *The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.*

Site-Specific Analysis

The Control Room is excluded from Table R-2 since it is included EAL RA3.1 for all modes. NEI 99-01 Rev. 6 developer notes state that the control room does not need to be included Table R-2 for this reason.

The Control Room is included in Table H-2 for all modes since adequate engineered safety/design features are not in place to preclude a Control Room evacuation due to the release of a hazardous gas at SSES.

The remaining list of mode dependent rooms in Tables R-2 and H-2 is a list of rooms where actions are absolutely required to be performed to move the plant from normal operations through cool down and to achieve and maintain cold shutdown. These areas are not areas requiring entry to solely meet surveillance requirements. This does not include actions called out for in the procedures that are not absolutely needed to move the plant into and maintain cold shutdown (e.g. shutting down turbine lube oils systems, opening of system drains). In order to transition from normal operations (Modes 1 and 2) to hot shutdown (Mode 3) the only location required is the Control Room since the plant is able to be placed into hot shutdown from the control room without the need for any other room entry.

The remaining rooms/areas shown in in Tables R-2 and H-2 were determined based on the discussion section above and Modes 1 and 2 were excluded. Therefore, the analysis starting point was a review the steps of GO-100/200-005 "PLANT SHUTDOWN TO HOT/COLD SHUTDOWN" after the Unit is has reached Mode 3. The analysis then branched to other procedure steps described in the GO's if those procedures steps were not excluded from consideration as noted in the discussion section. The analysis concluded that Reactor Building areas and elevations listed in in Tables R-2 and H-2 contain equipment that must be manipulated in Mode 3 to achieve and maintain cold shutdown. Modes 4 and 5 are also included for these areas to account for swapping loops for long term operation of shutdown cooling.

Tables R-2 and H-2 Bases

A review of station operating procedures identified the following mode dependent in-plant actions and associated areas that are required for normal plant operation, cooldown or shutdown

Unit 1 RA3.2 / HA5.1

Elevation	Area(s)	Mode(s)
670	27	3,4,5
683	27, 28 & 29	3,4,5
703	28 & 29	3,4,5
719	25 & 29	3,4,5
749	25 & 29	3,4,5
729 *	12, 21	1,2,3,4,5,D

Unit 2 RA3.2 / HA5.1

Elevation	Area(s)	Mode(s)
670	32	3,4,5
683	32, 33 & 34	3,4,5
703	33 & 34	3,4,5
719	30 & 34	3,4,5
749	32 & 33	3,4,5
729 *	12, 21	1,2,3,4,5,D

* Control Room – only applicable to HA5

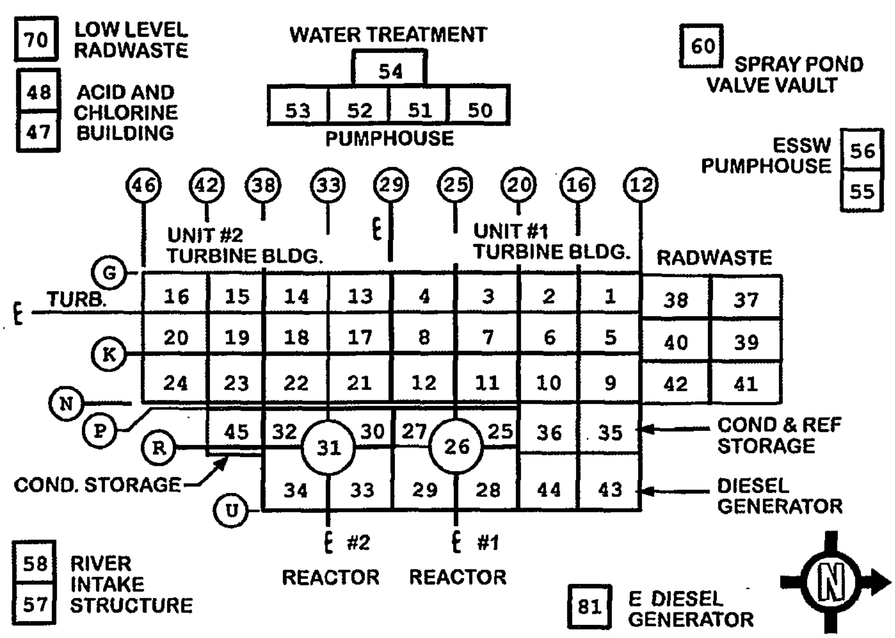
Table R-2 & H-2 Results

Table R-2 & H-2	Safe Operation & Shutdown Areas		
Elevation	**Unit 1 Area(s) ****	**Unit 2 Area(s) ****	**Mode(s)**
670'	RB 27	RB 32	3,4,5
683'	RB 27, 28, 29	RB 32, 33, 34	3,4,5
703'	RB 28, 29	RB 33, 34	3,4,5
719'	RB 25, 29	RB 30, 34	3,4,5
749'	RB 25, 29	RB 32, 33	3,4,5
729'	CS 12, 21*	CS 12, 21*	1, 2, 3, 4, 5, D

* Control Room – only applicable to HA5

** See Chart 1 for the specific locations of areas listed in Table R-2 and H-2.

Chart 1- Plant Area Key Plan



R Abnormal Rad Levels / Rad Effluent

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
1 Rad Effluent	IC Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem child thyroid CDE	IC Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem child thyroid CDE	IC Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem child thyroid CDE	IC Release of gaseous or liquid radioactivity greater than 2 times the TRM limits for 60 minutes or longer
	RG1.1 [1 2 3 4 5 D] Gaseous effluent > Table R-1 column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4) **RG1.2** [1 2 3 4 5 D] Dose assessment using actual meteorology indicates doses > 1,000 mrem TEDE or 5000 mrem child thyroid CDE at or beyond the EMERGENCY PLAN BOUNDARY (Notes 3, 4) **RG1.3** [1 2 3 4 5 D] Field survey results indicate EITHER of the following at or beyond the EMERGENCY PLAN BOUNDARY: • Closed window dose rates > 1,000 mR/hr expected to continue for ≥ 60 min. • Analyses of field survey samples indicate child thyroid CDE > 5,000 mrem for 60 min. of inhalation. (Notes 1, 2)	**RS1.1** [1 2 3 4 5 D] Gaseous effluent > Table R-1 column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4) **RS1.2** [1 2 3 4 5 D] Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem child thyroid CDE at or beyond the EMERGENCY PLAN BOUNDARY (Notes 3, 4) **RS1.3** [1 2 3 4 5 D] Field survey results indicate EITHER of the following at or beyond the EMERGENCY PLAN BOUNDARY: • Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min. • Analyses of field survey samples indicate child thyroid CDE > 500 mrem for 60 min. of inhalation. (Notes 1, 2)	**RA1.1** [1 2 3 4 5 D] Gaseous effluent > Table R-1 column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4) **RA1.2** [1 2 3 4 5 D] Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem child thyroid CDE at or beyond the EMERGENCY PLAN BOUNDARY (Notes 3, 4) **RA1.3** [1 2 3 4 5 D] Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem child thyroid CDE at or beyond the EMERGENCY PLAN BOUNDARY for 60 min. of exposure (Notes 1, 2) **RA1.4** [1 2 3 4 5 D] Field survey results indicate EITHER of the following at or beyond the EMERGENCY PLAN BOUNDARY: • Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min. • Analyses of field survey samples indicate child thyroid CDE > 50 mrem for 60 min. of inhalation. (Notes 1, 2)	**RU1.1** [1 2 3 4 5 D] Gaseous or liquid effluent > Table R-1 column "UE" for ≥ 60 min. (Notes 1, 2, 3) **RU1.2** [1 2 3 4 5 D] Sample analyses for a gaseous or liquid release indicates a concentration or release rate > 2 x TRM limits for ≥ 60 min. (Notes 1, 2)
2 Irradiated Fuel Event	IC Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer	IC Spent fuel pool level at the top of the fuel racks	IC Significant lowering of water level above or damage to irradiated fuel	IC Unplanned loss of water level above irradiated fuel
	RG2.1 [1 2 3 4 5 D] Spent fuel pool level CANNOT BE RESTORED to at least 0.5ft. Above the top of the spent fuel racks for ≥ 60 min. (Note 1)	**RS2.1** [1 2 3 4 5 D] Lowering of spent fuel pool level to ≤ 0.5 ft. above the top of the spent fuel racks	**RA2.1** [1 2 3 4 5 D] Uncovery of irradiated fuel in the REFUELING PATHWAY **RA2.2** [1 2 3 4 5 D] Damage to irradiated fuel resulting in a release of radioactivity **AND** Any of the following radiation monitor indications: • Refuel Floor High Exhaust (> 18 mR/hr) • Refuel Floor Wall Exhaust (> 21 mR/hr) • Channel 14 Spent Fuel Pool Area Criticality Monitor (> 100 mR/hr) • Channel 15 Refueling Floor Area (> 80 mR/hr) • Channel 42 Refueling Floor Area (> 80 mR/hr) • Channel 47 Unit 1/ 44 (Unit 2) Spent Fuel Pool Area Criticality Monitor (> 100 mR/hr) • Channel 49 Refueling Floor High Range Monitor (on scale) **RA2.3** [1 2 3 4 5 D] Lowering of spent fuel pool level to ≤ 10 ft. above the top of the spent fuel racks	**RU2.1** [1 2 3 4 5 D] UNPLANNED water level drop in the REFUELING PATHWAY as indicated by ANY of the following on EITHER unit: • Fuel Pool Water Low Level alarm • Skimmer Surge Tank Low Level alarm • Visual observation of a water level drop below a fuel pool skimmer surge tank inlet • Observation of water draining down the outside wall of primary containment **AND** UNPLANNED rise in area radiation levels as indicated by ANY of the following radiation monitors: • Channel 14 Spent Fuel Pool Area Criticality Monitor • Channel 15 Refueling Floor Area • Channel 42 Refueling Floor Area
3 Area Radiation Levels	None	None	IC: Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown, or shutdown **RA3.1** [1 2 3 4 5 D] Dose rate > 15 mR/hr in ANY of the following areas: • Main Control Room • Radwaste Control Room • Both the Central Alarm Station (CAS) and Secondary Alarm Station (SAS) **RA3.2** [3 4 5] An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to ANY Table R-2 rooms or areas (Note 5)	None

Notes

- **Note 1:** The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded
- **Note 2:** If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit
- **Note 3:** If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is NO longer VALID for classification purposes
- **Note 4:** The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available
- **Note 5:** 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

Table R-1 Effluent Monitor Classification Thresholds (Note 4)

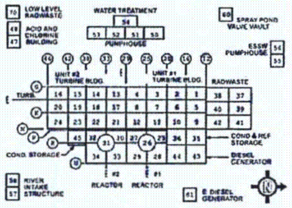
	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Plant Vent (noble gas)	0C630 0C667	1.9E+09 µCi/min (site total)	1.9E+08 µCi/min (site total)	1.9E+07 µCi/min (site total)	4.0E+06 µCi/min (site total)
Liquid	LRW	RR-06433	–	–	–	2 x hi alarm
Liquid	1(2) RHRSW A/B	RR-D12-1(2)1R606	–	–	–	2 x hi alarm
Liquid	1(2) SW/SDHR	RR-D12-1(2)R604	–	–	–	2 x hi alarm

Table R-2 Safe Operation & Shutdown Areas

Elevation	Unit 1 Area(s) **	Unit 2 Area(s) **	Mode(s)
670'	RB 27	RB 32	3, 4, 5
683'	RB 27, 28, 29	RB 32, 33, 34	3, 4, 5
703'	RB 28 29	RB 33, 34	3, 4, 5
719'	RB 25, 29	RB 30, 34	3, 4, 5
749'	RB 25, 29	RB 32, 33	3, 4, 5

** See Chart 1 for location of Plant Areas

Chart 1 – Plant Area Key Plan



C — Cold SD/ Refueling System Malfunct.

1 — RPV Level

GENERAL EMERGENCY

IC: Loss of RPV inventory affecting fuel clad integrity with Containment challenged

CG1.1 (4, 5)
RPV level < -161 in. (TAF) for ≥ 30 min. (Note 1)
AND
ANY Containment Challenge indication, Table C-2:

CG1.2 (4, 5)
RPV level cannot be monitored for ≥ 30 min. (Note 1)
AND
UNPLANNED increase in **ANY** Table C-1 sump or tank level due to a loss of RPV inventory of sufficient magnitude to indicate core uncovery
AND
ANY Containment Challenge indication, Table C-2

SITE AREA EMERGENCY

IC Loss of RPV inventory affecting core decay heat removal capability

CS1.1 (4, 5)
CONTAINMENT CLOSURE not established
AND
RPV level < -129 in. (Level 1)

CS1.2 (4, 5)
CONTAINMENT CLOSURE established
AND
RPV level < -161 in. (TAF)

CS1.3 (4, 5)
RPV level cannot be monitored for ≥ 30 min. (Note 1)
AND
UNPLANNED increase in **ANY** Table C-1 sump or tank level due to a loss of RPV inventory of sufficient magnitude to indicate core uncovery

ALERT

IC Loss of RPV inventory

CA1.1 (4, 5)
Loss of RPV inventory as indicated by RPV level < -38 in. (Level 2)

CA1.2 (4, 5)
RPV level cannot be monitored for ≥ 15 min. (Note 1)
AND
UNPLANNED increase in **ANY** Table C-1 sump or tank levels due to a loss of RPV inventory

UNUSUAL EVENT

IC Unplanned loss of RPV inventory for 15 minutes or longer

CU1.1 (4, 5)
UNPLANNED loss of reactor coolant results in RPV level less than a required lower limit ≥ 15 min.
(Note 1)

CU1.2 (4, 5)
RPV level cannot be monitored
AND
UNPLANNED increase in ANY Table C-1 sump or tank levels due to a loss of RPV inventory

2 — Loss of Essential AC Power

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT

IC Loss of **ALL** offsite power and **ALL** onsite AC power to essential buses for 15 minutes or longer

CA2.1 (4, 5, D)
Loss of **ALL** offsite and **ALL** onsite AC power capability to ALL 4.16 kV ESS buses on **EITHER** unit for ≥ 15 min. (Note 1)

UNUSUAL EVENT

IC Loss of **ALL** but one AC power source to essential buses for 15 minutes or longer

CU2.1 (4, 5, D)
AC power capability to **ALL** 4.16 kV ESS buses on **EITHER** unit reduced to a single power **source** for ≥ 15 min. (Note 1)
AND
ANY additional single power source failure will result in loss of ALL AC power to SAFETY SYSTEMS

3 — Loss of Vital DC Power

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT

IC Loss of Vital DC power for 15 minutes or longer

CU3.1 (4, 5)
< 105 VDC bus voltage indications on Technical Specification **REQUIRED** 125 VDC buses on the affected unit for ≥ 15 min. (Note 1)

4 — RCS Temp.

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT

IC Inability to maintain plant in cold shutdown

CA4.1 (4, 5)
UNPLANNED increase in RCS temperature to > 200°F for > Table C-3 duration (Note 1)
OR
UNPLANNED RPV pressure increase > 10 psig due to loss of decay heat removal capability

UNUSUAL EVENT

IC: UNPLANNED increase in RCS temperature

CU4.1 (4, 5)
UNPLANNED increase in RCS temperature to > 200°F due to loss of decay heat removal capability

CU4.2 (4, 5)
Loss of ALL RCS temperature and RPV level indication for ≥ 15 min. (Note 1)

5 — Loss of Comm.

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT

IC: Loss of **ALL** onsite or offsite communications capabilities

CU5.1 (4, 5, D)
Loss of ALL Table C-4 onsite communication methods
OR
Loss of ALL Table C-4 ORO communication methods
OR
Loss of ALL Table C-4 NRC communication methods

6 — Hazardous Events Affecting Safety Systems

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT

IC: Hazardous event affecting a SAFETY SYSTEM required for the current operating mode

CA6.1 (4, 5)
The occurrence of **any Table** C-5 hazardous event
AND
EITHER:
Event damage has caused indications of degraded performance in a least one train of a SAFETY SYSTEM required for the current operating mode.
OR
The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure required for the current operating mode

UNUSUAL EVENT: None

Notes

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is **NOT** required.

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Table C-1 Sumps & Tanks

- Drywell equipment drain tank
- Drywell sumps
- Reactor Building sump
- LRW collection tanks
- Main condenser hotwell
- Suppression pool
- Visual observation

Table C-2 Containment Challenge Indiations

- CONTAINMENT CLOSURE **not** established (Note 6)
- PC hydrogen concentration > 6%
- UNPLANNED rise in PC pressure
- Exceeding one or more Secondary Containment Control Max Safe Radiation Levels (EO-000-104 Table 9) that can be read in the Control Room (Table C-6)

Table C-3 RCS Reheat Duration Thresholds

RCS Status	CONTAINMENT CLOSURE Status	Heat-up Duration
INTACT	N/A	60 min. *
Not INTACT	Established	20 min. *
	NOT Established	0 min.

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

Table C-4 Communication Methods

System	Onsite	ORO	NRC
• UHF Radio	X		
• Plant PA System	X		
• Dedicated Conference Lines		X	
• Commercial Telephone Systems	X	X	X
• Cellular Telephone		X	X
• FTS-2001 (ENS)		X	X

Table C-5 Hazardous Events

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

Table C-6 Max Safe Reactor Building Radiation Limits

RB AREA Elevation (ft.)	ARM Number	ARM Channel Description	Max Safe Rad Limit (R/HR)
818	49	Refuel Floor	10
749	52 54	RWCU Recirc PP Access Fuel Pool PP Area	10
719	50 51	CRD North CRD South	10
670	53	Remote Shutdown Room	10
645	48 57 55 56	HPCI PP & Turbine Room RCIC PP & Turbine Room RHR A C PP Room RHR B D PP Room	10

Modes: 1 Power Operations | 2 Startup | 3 Hot Shutdown | 4 Cold Shutdown | 5 Refueling | D Defueled

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COLD CONDITIONS (RCS ≤ 200°F)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

H Hazards

1 Security

GENERAL EMERGENCY

IC: HOSTILE ACTION resulting in loss of physical control of the facility

HG1.1 [1 | 2 | 3 | 4 | 5 | D]

A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision
AND
EITHER of the following has occurred:
Any of the following safety functions cannot be controlled or maintained
- Reactivity
- RPV water level
- RCS heat removal

OR
Damage to spent fuel has occurred or is IMMINENT

SITE AREA EMERGENCY

IC: HOSTILE ACTION within the PROTECTED AREA

HS1.1 [1 | 2 | 3 | 4 | 5 | D]

A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision

ALERT

IC: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

HA1.1 [1 | 2 | 3 | 4 | 5 | D]

A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervision
OR
A validated notification from NRC of an aircraft attack threat within 30 min. of the site

UNUSUAL EVENT

IC: Confirmed SECURITY CONDITION or threat

HU1.1 [1 | 2 | 3 | 4 | 5 | D]

A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the Security Shift Supervision
OR
Notification of a credible security threat directed at the site
OR
A validated notification from the NRC providing information of an aircraft threat

2 Seismic Event

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT

IC: Seismic event greater than OBE levels

HU2.1 [1 | 2 | 3 | 4 | 5 | D]

Seismic event greater than OPERATING BASIS EARTHQUAKE (OBE) as indicated by seismic instrumentation in the Control Room recording level greater than an OBE

3 Natural or Tech. Hazard

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT

IC: Hazardous event

HU3.1 [1 | 2 | 3 | 4 | 5 | D]

A tornado strike within the PROTECTED AREA

HU3.2 [1 | 2 | 3 | 4 | 5 | D]

Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component required for the current operating mode

HU3.3 [1 | 2 | 3 | 4 | 5 | D]

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)

HU3.4 [1 | 2 | 3 | 4 | 5 | D]

A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)

4 Fire

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT: None

UNUSUAL EVENT

IC: FIRE potentially degrading the level of safety of the plant

HU4.1 [1 | 2 | 3 | 4 | 5 | D]

A FIRE is NOT extinguished within 15 min. of ANY of the following FIRE detection indications (Note 1):
- 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
The FIRE is located within ANY Table H-1 area

HU4.2 [1 | 2 | 3 | 4 | 5 | D]

Receipt of a single fire alarm (i.e., no other indications of a FIRE)
AND
The fire alarm is indicating a FIRE within ANY Table H-1 area
AND
The existence of a FIRE is NOT verified (i.e. proved or disproved) within 30 min. of alarm receipt (Note 1)

HU4.3 [1 | 2 | 3 | 4 | 5 | D]

A FIRE within the plant PROTECTED AREA NOT extinguished within 60 min. of the initial report, alarm or indication (Note 1)

HU4.4 [1 | 2 | 3 | 4 | 5 | D]

A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

5 Hazardous Gases

GENERAL EMERGENCY: None

SITE AREA EMERGENCY: None

ALERT

IC: Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

HA5.1 [1 | 2 | 3 | 4 | 5 | D]

Release of a toxic, corrosive, asphyxiant or flammable gas into any Table H-2 rooms or areas
AND
Entry into the room or area is prohibited or IMPEDED (Note 5)

UNUSUAL EVENT: None

6 Control Room Evacuation

GENERAL EMERGENCY: None

SITE AREA EMERGENCY

IC: Inability to control a key safety function from outside the Control Room

HS6.1 [1 | 2 | 3 | 4 | 5 | D]

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels
AND
Control of any of the following key safety functions is not reestablished within 15 min. (Note 1):
- Reactivity
- RPV water level
- RCS heat removal

ALERT

IC: Control Room evacuation resulting in transfer of plant control to alternate locations

HA6.1 [1 | 2 | 3 | 4 | 5 | D]

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels

UNUSUAL EVENT: None

7 ED/RM Judgment

GENERAL EMERGENCY

IC: Other conditions existing that in the judgment of the Emergency Director/Recovery Manager warrant declaration of General Emergency

HG7.1 [1 | 2 | 3 | 4 | 5 | D]

Other conditions exist which in the judgment of the Emergency Director/Recovery Manager 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.

SITE AREA EMERGENCY

IC: Other conditions existing that in the judgment of the Emergency Director/Recovery Manager warrant declaration of Site Area Emergency

HS7.1 [1 | 2 | 3 | 4 | 5 | D]

Other conditions exist which in the judgment of the Emergency Director/Recovery Manager 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 EMERGENCY PLAN BOUNDARY.

ALERT

IC: Other conditions existing that in the judgment of the Emergency Director/Recovery Manager warrant declaration of an Alert

HA7.1 [1 | 2 | 3 | 4 | 5 | D]

Other conditions exist which in the judgment of the Emergency Director/Recovery Manager 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.

UNUSUAL EVENT

IC: Other conditions existing that in the judgment of the Emergency Director/Recovery Manager warrant declaration of a UE

HU7.1 [1 | 2 | 3 | 4 | 5 | D]

Other conditions exist which in the judgment of the Emergency Director/Recovery Manager 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

Notes

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded

Note 5: 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

Note 7: This EAL does NOT apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents

Table H-1 Fire Areas

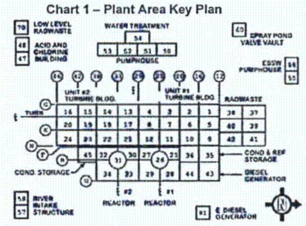
- Control Structure
- Diesel Generator Buildings
- ESSW Pump House
- Reactor Buildings
- Turbine Buildings
- ISFSI

Table H-2 Safe Operation & Shutdown Areas

Elevation	Unit 1 Area(s)**	Unit 2 Area(s)**	Mode(s)
670'	RB 27	RB 32	3, 4, 5
683'	RB 27, 28, 29	RB 32, 33, 34	3, 4, 5
703'	RB 28, 29	RB 33, 34	3, 4, 5
719'	RB 25, 29	RB 30, 34	3, 4, 5
749'	RB 25, 29	RB 32, 33	3, 4, 5
729'	CS 12, 21	CS 12, 21	1, 2, 3, 4, 5, D

** See Chart 1 for location of Plant Areas

Chart 1 – Plant Area Key Plan



Modes: [1] Power Operations [2] Startup [3] Hot Shutdown [4] Cold Shutdown [5] Refueling [D] Defueled

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Form EP-RM-004-H

ALL CONDITIONS

S – System Malfunct.

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
1 Loss of Essential AC Power	**IC: Prolonged loss of ALL offsite and all onsite AC power to essential buses** **SG1.1** [1] [2] [3] Loss of ALL offsite and ALL onsite AC power capability to ALL 4.16 kV ESS buses on EITHER unit AND EITHER: Restoration of at least one 4.16 kV ESS bus in < 4 hours is NOT likely (Note 1) OR RPV water level CANNOT BE RESTORED AND MAINTAINED > -179 in.	**IC: Loss of ALL offsite and all onsite AC power to essential buses for 15 minutes or longer** **SS1.1** [1] [2] [3] Loss of ALL offsite and ALL onsite AC power capability to ALL 4.16 kV ESS buses on EITHER unit for ≥ 15 min. (Note 1)	**IC: Loss of all but one AC power source to essential buses or 15 minutes or longer** **SA1.1** [1] [2] [3] AC power capability to ALL 4.16 kV ESS buses on EITHER unit reduced to a single power source for ≥ 15 min. (Note 1) AND ANY additional single power source failure will result in loss of ALL AC power to SAFETY SYSTEMS	**IC: Loss of ALL offsite AC power capability to essential buses or 15 minutes or longer** **SU1.1** [1] [2] [3] Loss of ALL offsite AC power capability to ALL 4.16 kV ESS buses on EITHER unit for ≥ 15 min. (Note 1)
2 Loss of Vital DC Power	**IC: Loss of ALL essential AC and vital DC power sources for 15 minutes or longer** **SG1.2** [1] [2] [3] Loss of ALL offsite and ALL onsite AC power capability to ALL 4.16 kV ESS buses on EITHER unit for ≥ 15 min. AND Indicated voltage is < 105 VDC on ALL of the following vital 125 VDC main distribution buses on the affected unit for ≥ 15 min. (Note 1) • 1D612 (2D612) • 1D622 (2D622) • 1D632 (2D632) • 1D642 (2D642)	**IC: Loss of ALL vital DC power for 15 minutes or longer** **SS2.1** [1] [2] [3] Indicated voltage is < 105 VDC on ALL of the following vital 125 VDC main distribution buses on the affected unit for ≥ 15 min. (Note 1): • 1D612 (2D612) • 1D622 (2D622) • 1D632 (2D632) • 1D642 (2D642)	None	None
3 Loss of Control Room Indications	None	None	**IC: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress** **SA3.1** [1] [2] [3] An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1) AND ANY significant transient is in progress, Table S-2	**IC: UNPLANNED loss of Control Room indications for 15 minutes or longer** **SU3.1** [1] [2] [3] An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1)
4 RCS Activity	None	None	None	**IC: Reactor coolant activity greater than Technical Specification allowable limits** **SU4.1** [1] [2] [3] Offgas pretreatment monitor high-high radiation alarm **SU4.2** [1] [2] [3] Coolant activity > 0.2 μCi/gm dose equivalent I-131 for > 48 hours OR Coolant activity > 4.0 μCi/gm dose equivalent I-131 at any time
5 RCS Leakage	None	None	None	**IC: RCS leakage for 15 minutes or longer** **SU5.1** [1] [2] [3] RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. OR RCS identified leakage > 25 gpm for ≥ 15 min. OR Leakage from the RCS to a location outside Primary Containment > 25 gpm for ≥ 15 min. (Note 1)
6 RPS Failure	None	**IC: Inability to shut down the reactor causing a challenge to RPV water level or RCS heat removal** **SS6.1** [1] [2] An automatic or manual scram fails to shut down the reactor AND ALL actions to shut down the reactor are NOT successful as indicated by reactor power ≥ 5% AND EITHER: RPV level CANNOT BE RESTORED AND MAINTAINED > -179 in. or CANNOT be determined OR Suppression pool water temperature AND RPV pressure CANNOT BE MAINTAINED below the Heat Capacity Temperature Limit (Figure – HCTL)	**IC: Automatic or manual scram fails to shut down the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor** **SA6.1** [1] [2] An automatic or manual scram fails to shut down the reactor AND Manual scram actions taken at the reactor control console (Manual PBs, Mode Switch, ARI) are NOT successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8)	**IC: Automatic or manual scram fails to shut down the reactor** **SU6.1** [1] [2] An automatic scram did not shut down the reactor after ANY RPS setpoint is exceeded AND A subsequent automatic scram or manual scram action taken at the reactor control console (Manual PBs, Mode Switch, ARI) is successful in shutting down the reactor as indicated by reactor power < 5% (APRM downscale) (Note 8) **SU6.2** [1] [2] A manual scram did NOT shutdown the reactor after ANY manual scram action was initiated AND A subsequent automatic scram or manual scram action taken at the reactor control console (Manual PBs, Mode Switch, ARI) is successful in shutting down the reactor as indicated by reactor power < 5% (APRM downscale) (Note 8)
7 Loss of Comm.	None	None	None	**IC: Loss of ALL onsite or offsite communications capabilities** **SU7.1** [1] [2] [3] Loss of ALL Table S-3 onsite communication methods OR Loss of ALL Table S-3 ORO communication methods OR Loss of ALL Table S-3 NRC communication methods
8 Hazardous Event Affecting Safety Systems	None	None	**IC: Hazardous event affecting a SAFETY SYSTEM required for the current operating mode** **SA8.1** [1] [2] [3] The occurrence of ANY Table S-4 hazardous event AND EITHER: Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM required for the current operating mode OR The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure required for the current operating mode	None

Notes

Note 1: The ED/RM should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded

Note 8: A manual scram 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

Table S-1 Safety System Parameters

- Reactor power
- RPV water level
- RPV pressure
- Primary Containment pressure
- Suppression Pool water level
- Suppression Pool temperature

Table S-2 Significant Transients

- Reactor scram
- Runback > 25% reactor power
- RRC pump trip while > 25% reactor power
- ECCS injection
- Thermal power oscillations > 10%

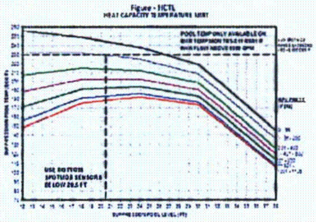
Table S-3 Communication Methods

System	Onsite	ORO	NRC
• UHF Radio	X		
• Plant PA System	X		
• Dedicated Conference Lines		X	
• Commercial Telephone Systems	X	X	X
• Cellular Telephone		X	X
• FTS-2001 (ENS)		X	X

Table S-4 Hazardous Events

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

Figure - HCTL



Modes: 1 Power Operations | 2 Startup | 3 Hot Shutdown | 4 Cold Shutdown | 5 Refueling | D Defueled

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Form EP-RM-004-S

HOT CONDITIONS (RCS > 200°F)

		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
E ISFSI	**1** Confinement Boundary	None	None	None	**IC: Damage to a loaded cask CONFINEMENT BOUNDARY** **EU1.1** 1 2 3 4 5 D Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by a radiation reading on a loaded spent fuel cask > **ANY** of the following: • 800 mrem/hr at 3 ft from the HSM surface • 200 mrem/hr on contact on the outside of the HSM door centerline • 40 mrem/hr on contact on the end shield wall exterior

Modes: 1 Power Operations, 2 Startup, 3 Hot Shutdown, 4 Cold Shutdown, 5 Refueling, D Defueled	Susquehanna Nuclear, LLC	Form EP-RM-004-E ALL CONDITIONS

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
F Fission Product Barrier Degradation	IC: Loss of ANY two barriers AND loss of potential loss of the third barrier. FG1.1 1 2 3 Loss of ANY two barriers AND Loss or potential loss of the third barrier (Table F-1)	IC: Loss or potential loss of ANY two barriers FS1.1 1 2 3 Loss or potential loss of ANY two barriers (Table F-1)	IC: ANY loss or ANY potential loss either fuel clad or RCS FA1.1 1 2 3 ANY loss or ANY potential loss of EITHER Fuel Clad or RCS barrier (Table F-1)	None

Table F-1 Fission Product Barrier Threshold Matrix

	FC - Fuel Clad Barrier		RCS - Reactor Coolant System Barrier		PC - Primary Containment Barrier	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A RPV Level	1. SAGs entered	1. RPV level CANNOT BE RESTORED AND MAINTAINED > -161 in. (TAF) or CANNOT be determined.	1. RPV level CANNOT BE RESTORED AND MAINTAINED > -161 in. (TAF) or CANNOT be determined	None	None	1. SAGs entered
B RCS Leak Rate	None	None	1. UNISOLABLE break in ANY of the following: • Main Steam Line • HPCI Steam Line • RCIC Steam Line • RWCU • Feedwater OR 2. Emergency RPV Depressurization is required	1. UNISOLABLE primary system leakage that results in exceeding EITHER: One or more Max Normal Reactor Building Radiation Limits (EO-000-104 Table 8) that can be read in the Control Room (Table F-3) OR One or more Max Normal Reactor Building area temperature Limits (EO-000-104 Table 8) that can be read at the Control Room (Table F-4)	1. UNISOLABLE primary system leakage that results in exceeding EITHER: One or more Max Safe Reactor Building Radiation Limits (EO-000-104 Table F-8) that can be read in the Control Room (Table F-5) OR One or more Max Safe Reactor Building area temperature Limits (EO-000-104 Table 8) that can be read in the Control Room (Table F-6)	None
C PC Conditions	None	None	1. Primary Containment pressure > 1.72 psig due to RCS leakage	None	1. UNPLANNED rapid drop in Primary Containment pressure following Primary Containment pressure rise OR 2. Primary Containment pressure response NOT consistent with LOCA conditions	1. Primary Containment Pressure > 53 psig OR 2. Deflagration concentrations exist inside PC ($H_2 \geq 6\%$ AND $O_2 \geq 5\%$) OR 3. Heat Capacity Temperature Limit (Figure – HCTL) exceeded
D PC Radiation / RCS Activity	1. CHRRM radiation > 3.0E+3 R/hr OR 2. Primary coolant activity > 300 µCi/gm I-131 dose equivalent	None	1. CHRRM radiation > 7.0E+0 R/hr with indication of a RCS leak inside the drywell	None	None	1. CHRRM radiation > 4.0E+4 R/hr
E PC Integrity or Bypass	None	None	None	None	1. UNISOLABLE direct downstream pathway to the environment exists after Primary Containment isolation signal OR 2. Intentional Primary Containment venting per EP-DS-004 RPV and PC Venting	None
F ED/RM Judgment	1. ANY condition in the opinion of the Emergency Director/Recovery Manager that indicates loss of the fuel clad barrier	1. ANY condition in the opinion of the Emergency Director/Recovery Manager that indicates potential loss of the Fuel Clad barrier	1. ANY condition in the opinion of the Emergency Director/Recovery Manager that indicates loss of the RCS barrier	1. ANY condition in the opinion of the Emergency Director/Recovery Manager that indicates potential loss of the RCS barrier	1. ANY condition in the opinion of the Emergency Director/Recovery Manager that indicates loss of the Primary Containment barrier	1. ANY condition in the opinion of the Emergency Director/Recovery Manager that indicates potential loss of the Primary Containment barrier

Table F-2 Fission Product Barrier Status Table

Circle the X's in the table for all applicable situations. Declare the EAL based upon all circled X's in any column.	FG1.1: General Emergency Loss of ANY two barriers and loss or potential loss of third barrier				FS1.1: Site Area Emergency Loss or potential loss of ANY two barriers												FA1.1: Alert ANY loss OR ANY potential loss of either Fuel Clad OR RCS Barrier			
Fuel Clad - Loss	X	X		X	X		X		X	X							X			
Fuel Clad – Potential Loss			X			X		X			X	X						X		
RCS – Loss	X	X	X		X			X					X	X					X	
RCS – Potential Loss				X		X	X								X	X				X
Primary Containment - Loss	X		X	X					X		X		X		X					
Primary Containment – Potential Loss		X								X		X		X		X				

Figure - HCTL
HEAT CAPACITY TEMPERATURE LIMIT

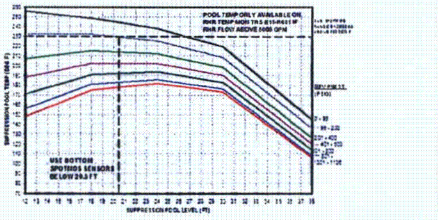

Table F-3
Max Normal Reactor Building Radiation Limits

RB AREA Elevation (ft.)	ARM Number	ARM Channel Description	Max Norm Rad Limit
818	35 14 15 42 47 (44 U2)	Cask Stor Area Spent Fuel Crit Mon Refuel Floor North (South U2) Refuel Floor West Spent Fuel Crit Mon	Hi Alarm
749	8 10 11	RWCU Recirc PP Access Fuel Pool PP Area Rx Bldg Sample St	Hi Alarm
719	5 6	CRD North CRD South	Hi Alarm
670	18	Remote Shutdown Room	Hi Alarm
645	3 2 25 1 4	HPCI PP & Turbine Room RCIC PP & Turbine Room RHR A C PP Room RHR B D PP Room RB/RW Sump Area	Hi Alarm

Table F-4
Max Normal Reactor Building Temperature Limits

RB AREA Elevation (ft.)	Area Temperature	Max Norm Temp (°F)
749	RWCU – Pump Room RWCU – Heat Exch Room RWCU – Penetration Room	120 120 120
719	Main Steam Line Tunnel	157
683	HPCI Pipe Routing Area RCIC Pipe Routing Area	120 120
645	HPCI – Equip Area HPCI – Emerg Area Cooler	120 120
645	RCIC – Emerg Area Cooler RCIC – Equip Area	120 120
645	RHR Equip Area 1	110
645	RHR Equip Area 2	110

Table F-5
Max Safe Reactor Building Radiation Limits

RB AREA Elevation (ft.)	ARM Number	ARM Channel Description	Max Safe Rad Limit (R/HR)
818	49	Refuel Floor	10
749	52 54	RWCU Recirc PP Access Fuel Pool PP Area	10
719	50 51	CRD North CRD South	10
670	53	Remote Shutdown Room	10
645	48 57 55 56	HPCI PP & Turbine Room RCIC PP & Turbine Room RHR A C PP Room RHR B D PP Room	10

Table F-6
Max Safe Reactor Building Temperature Limits

RB Area Elevation (ft.)	Area Temperature	Max Safe Temp (°F)
749	RWCU – Pump Room RWCU – Heat Exch Room RWCU – Penetration Room	147 147 131
719	Main Steam Line Tunnel	177
683	HPCI Pipe Routing Area RCIC Pipe Routing Area	167 167
645	HPCI – Equip Area HPCI – Emerg Area Cooler	167 167
645	RCIC – Emerg Area Cooler RCIC – Equip Area	167 167
645	RHR Equip Area 1	142
645	RHR Equip Area 2	142

Modes:	1	2	3	4	5	D	Susquehanna Nuclear, LLC	Form EP-RM-004-F
	Power Operations	Startup	Hot Shutdown	Cold Shutdown	Refueling	Defueled		HOT CONDITIONS (RCS > 200°F)