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DTE Energy

July 28, 2015
NRC-15-0077

10 CFR 50.90

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

References: 1) Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

2) DTE Electric Company Letter to the NRC, "License Amendment Request to Revise the Emergency Action Level Scheme for the Emergency Plan," NRC-14-0055, dated October 21, 2014 (ML14295A078)

3) NRC Email to DTE Electric Company, "Fermi 2 - Request for Additional Information Regarding the License Amendment Request to Adopt NEI 99-01, Revision 6 (MF5048)", dated May 6, 2015 (ML15127A212)

4) DTE Electric Company Letter to the NRC, "Response to Request for Additional Information (RAI) Regarding the License Amendment Request to Revise the Emergency Action Level (EAL) Scheme for the Emergency Plan," NRC-15-0061, dated June 18, 2015 (ML15170A324)

5) NRC Email to DTE Electric Company, "Fermi 2 - Request for Additional Information Regarding the License Amendment Request to Adopt NEI 99-01, Revision 6 (MF5048)", dated July 6, 2015 (ML15191A057)

Subject: Response to Request for Additional Information (RAI) Regarding the License Amendment Request to Revise the Emergency Action Level (EAL) Scheme for the Emergency Plan

In Reference 2, DTE Electric Company (DTE) submitted a license amendment request (LAR) for Fermi 2 to revise the Emergency Action Level (EAL) scheme for the Emergency Plan consistent with the scheme described in Nuclear Energy Institute (NEI) 99-01, Revision 6.

In an NRC electronic mail message dated May 6, 2015 (Reference 3), the NRC requested additional information to support its continued review of the LAR in Reference 2.

By letter dated June 18, 2015 (Reference 4), DTE provided a response to the NRC's request for additional information (RAI) in Reference 3.

Subsequently, in an electronic mail message dated July 6, 2015 (Reference 5), the NRC issued an additional request for information. Reference 5 identified four additional questions pertaining to the proposed EALs, which were discussed in a teleconference on July 6, 2015.

Enclosure 1 provides DTE's response to the NRC staff RAI detailed in Reference 5. Enclosure 2 provides a revised EAL Technical Bases clean pages document that incorporates changes discussed in Enclosure 1.

This letter contains no new regulatory commitments.

Should you have any questions or require additional information, please contact Mr. Christopher R. Robinson of my staff at (734) 586-5076.

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

Executed on July 28, 2015

Vito A. Kaminskas
Site Vice President

Enclosures:

1) Response to Request for Additional Information (RAI) Regarding the License Amendment Request to Revise the Emergency Action Level (EAL) Scheme for the Emergency Plan
2) Revised (Clean) Fermi 2 EAL Technical Bases

cc: NRC Project Manager
NRC Resident Office
Reactor Projects Chief, Branch 5, Region III
Regional Administrator, Region III
Michigan Public Service Commission
Regulated Energy Division (kindschl@michigan.gov)

Enclosure 1 to
NRC-15-0077

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

Response to Request for Additional Information (RAI) Regarding the License Amendment Request to Revise the Emergency Action Level (EAL) Scheme for the Emergency Plan

1. ***<u>Follow-up to RAI #4 Response</u>:*** *DTE states that it has changed the established set of Max Safe instrumentation/values with an alternate table with appropriate threshold values. This did not appear to have changed with EALs CG1.1 and CG1.2. Please explain this apparent discrepancy or revise the values used for Max Safe radiation levels consistently throughout the entire EAL scheme.*

   Response

   EALs CG1.1 and CG1.2 have been revised to be consistent with the entire EAL scheme. These revised EALs only include the area radiation monitor (ARM) that supports the Max Safe radiation level (Reactor Building Sub-Basement (RBSB) Torus Room – ARM Channel 14).

   See response to question 2 below for additional information.

2. ***<u>Follow-up to RAI #4 and #29 Responses</u>:*** *The newly-developed Table F-2 uses instrumentation values that are significantly below the guidance of Max Safe radiation levels referenced in NEI 99-01, Revision 6. While EOP Table 14 lists all Max Safe levels at 5 R/hr, Table F-2 lists the EAL threshold values as 950 mR/hr for all secondary containment radiation monitors with a maximum detector scale of 1 R/hr, and does not address the omission of the detectors with a maximum scale of 100mr/hr. As proposed, DTE could potentially be declaring a General Emergency when secondary containment radiation levels are approximately 20% of the approved guidance's threshold value, resulting in a significant event over-classification and unwarranted protective actions being recommended for the general public. For all instances of Max Safe radiation values, please provide an adequate indication of Containment Challenge or Primary Containment Isolation Failure, as appropriate, with justification of how these indications are appropriate for the EAL scheme and meet the intent of the approved guidance.*

   Response

   Table F-2 has been deleted and the only radiation monitor that reaches the Max Safe radiation level (RBSB Torus Room – ARM Channel 14) is now included in EAL PC-Loss 2.A. As previously mentioned in the response to question 1, EALs CG1.1 and CG1.2 were also revised to only include the same radiation monitor. In addition, Table F-1, Fission Product Barrier Threshold Matrix, was revised accordingly.

   Due to the declaration timing requirement, only the installed monitoring instruments with a range that supports Max Safe radiation and temperature levels and can be read from the Control Room or immediately adjacent areas are included. The installed monitoring instruments that require a survey to determine Max Safe values are not included. ARMs in the current Fermi 2 EALs that would require a radiation survey to verify the Max Safe radiation values are bounded in classification consideration by the installed EOP Max Safe temperature instrumentation, and Primary Containment LOSS 5.A (UNISOLABLE direct downstream pathway to the environment exists after primary containment isolation signal).

This is a deviation from the guidance specified in NEI 99-01, Rev. 6, due to the inability to use other ARMs with range below the Max Safe radiation values.

3. ***Follow-up to RAI #6 Response:*** *DTE did not address the question regarding the difference between the currently-approved radiation level threshold values and the values included in the LAR, especially considering the guidance for these EALs is the same in their current scheme and in NEI 99-01, Revision 6. Please explain the difference between the currently-approved radiation level threshold values and the values included in the LAR and provide justification why the currently-used EAL threshold values are appropriate for the approved scheme.*

Response

For Division 1 of the Standby Gas Treatment (SBGT) System, the effluent monitor classification threshold values for the proposed EALs (RS1.1 and RG1.1) are the same as the values currently listed in EP-101 (AS1 and AG1). For Division 2 of the SBGT System, there is a slight difference between the proposed EALs (RS1.1 and RG1.1) and the values currently listed in EP-101 (AS1 and AG1). The proposed values are based on a measured flow rate and calibration coefficient for each Division of SBGT instead of representative values that were used for both divisions of SBGT in the current EALs.

The gaseous effluent monitor classification threshold values for the proposed EALs (RU1.1) are different than the existing EALs (AU1) due to a difference in source terms. The existing AU1 EAL utilized a source term that was derived from actual samples of gaseous effluents taken onsite. The proposed RU1.1 EAL uses a source term listed in UFSAR Chapter 11, Appendix A, Table IV-1. As stated in calculation EP-EALCALC-FERMI-1401 (Reference 4, Enclosure 4), utilizing the values in the UFSAR table provides an appropriate source term basis without requiring frequent validation with more recent sample analyses results.

The liquid effluent monitor classification threshold value for the proposed RU1.1 EAL is different than the existing AU1 EAL. While investigating the basis for the difference, it was determined that the existing value listed in EP-101 was incorrectly derived due to the inclusion of safety factors that should have been removed when determining the 2X ODCM limit, per the NUMARC guidance used at the time. This issue was entered into the Corrective Action Program and EP-101 has been revised. The value for the revised AU1 is now the same as the proposed RU1.1. The AA1 liquid effluent monitor classification threshold value in EP-101 was also impacted by this error and has been revised accordingly.

With the exception of the incorrect liquid effluent threshold value identified above, which has been corrected, the EP-101 EAL threshold values discussed above were verified to be appropriate for the approved scheme.

4. ***General Comment*:** *Based on the responses to the questions above, please provide a complete and clean (i.e., no markups) copy of your final EAL Basis Document.*

Response

The clean final EAL Technical Bases document is provided in Enclosure 2.

Enclosure 2 to
NRC-15-0077

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

Revised (Clean) Emergency Action Level Technical Bases

to be incorporated into
Implementing Procedure EP-101, “Classification of Emergencies”

TABLE OF CONTENTS

TABLE OF CONTENTS

1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for the Enrico Fermi Unit 2 Power Plant (Fermi 2). It should be used to facilitate review of the Fermi 2 EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EP-101, “Classification of Emergencies,” may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Director in making classifications, particularly those involving judgment or multiple events. The basis 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 in all cases of conditions present. Use of this document for assistance is not intended to delay the emergency classification.

Because the information in a basis document can affect emergency classification decision-making (e.g., the Emergency Director refers to it during an event), the NRC staff expects that changes to the basis document will be evaluated in accordance with the provisions of 10 CFR 50.54(q).

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 Fermi 2 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 ML13091A209) (Ref. 4.1.1), Fermi 2 conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barrier Thresholds

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. 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. Primary Containment (PC): The Primary Containment Barrier includes the drywell, the suppression pool, 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:

Alert:

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

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

2.4 EAL Organization

The Fermi 2 EAL scheme includes the following features:

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

  - EALs applicable only under cold 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 Fermi 2 EAL categories are aligned to and represent the NEI 99-01"Recognition Categories."
  - Subcategories are used in the Fermi 2 scheme as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The Fermi 2 EAL categories and subcategories are listed below.

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
R – Abnormal **R**ad Levels / Rad Effluent	1 – Radiological Effluent 2 – **I**rradiated 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 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 – **F**ission Product Barrier Degradation	None
Cold Conditions:	
C – **C**old Shutdown / Refueling System Malfunction	1 – RPV Level 2 – Loss of Essential AC Power 3 – RCS Temperature 4 – Loss of Vital DC Power 5 – Loss of Communications 6 – Hazardous Event Affecting Safety Systems

The primary tool for determining the emergency classification level is the EAL Classification Matrix. The user of the EAL Classification Matrix may (but is not required to) consult the EAL Technical Bases Document in order to obtain additional information concerning the EALs under classification consideration. The user should consult Section 3.0 and Attachments 1 & 2 of this document for such information.

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:

Category Letter & Title

Subcategory Number & Title

Initiating Condition (IC)

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

EAL Identifier (enclosed in rectangle)

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).

Classification (enclosed in rectangle):

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

EAL (enclosed in rectangle)

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

Mode Applicability

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

Definitions:

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.

Basis:

A Plant-Specific basis section that provides Fermi-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.

Fermi Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.6 Operating Mode Applicability (Ref. 4.1.7)

1 Power Operation

Mode Switch in Run

2 Startup

Mode Switch in Startup/Hot Standby or Mode Switch Refuel with all reactor vessel head closure bolts fully tensioned.

3 Hot Shutdown

Mode Switch in Shutdown and RCS temperature > 200°F

4 Cold Shutdown

Mode Switch in Shutdown and RCS temperature ≤ 200°F.

5 Refueling

Mode Switch in Shutdown or Refuel with one or more reactor vessel head closure bolts less then fully tensioned.

D Defueled

All nuclear fuel removed from reactor vessel (i.e., full core off load during refueling or extended outage).

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 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. 4.1.12).

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. The validation of indications should be completed in a manner that supports timely emergency declaration.

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.

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 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 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 with the ability to classify events and conditions based upon judgment using EALs that are consistent with the Emergency Classification Level (ECL) definitions (refer to Category H). The Emergency Director will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated 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. 4.1.14).

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 two different units, a Site Area Emergency should be declared.

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

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

Related guidance concerning classification of rapidly escalating events or conditions is provided in Regulatory Issue Summary (RIS) 2007-02, *Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events* (Ref. 4.1.2).

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

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. 4.1.2).

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.

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

<u>EAL momentarily met but the condition is corrected prior to an emergency declaration</u> – If an operator takes prompt manual action to address a condition, and the action is successful in correcting the condition prior to the emergency declaration, then the

applicable EAL is not considered met and the associated emergency declaration is not required. For illustrative purposes, consider the following example:

> An ATWS occurs and the 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 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. 4.1.3) 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. 4.1.3).

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 ML13091A209

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

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

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

4.1.5 10 § CFR 50.73 License Event Report System

4.1.6 Fermi 2 Offsite Dose Calculation Manual (ODCM) TRM Volume II Figure 3.0-1 Map Defining Unrestricted Areas and Site Boundary for Radioactive Gaseous and Liquid Effluents

4.1.7 Technical Specifications Table 1.1-1 Modes

4.1.8 EP-101 Classification of Emergencies, Rev. 39

4.1.9 Technical Specifications Section 3.6 Containment Systems

4.1.10 UFSAR Figure 1.2-5 Site Plot Plan

4.1.11 WG-001 Fermi Writers Guide

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

4.1.13 Fermi Certificate of Compliance No. 114 Appendix A Technical Specifications for the HI-STORM 100 Cask System

4.2 Implementing

4.2.1 EP-101Classification of Emergencies

4.2.2 NEI 99-01 Rev. 6 to Fermi EAL Comparison Matrix

4.2.3 Fermi EAL Matrix

4.3 Commitments

4.3.1 None

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (Ref. 4.1.1 except as noted)

Selected 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.

Alert

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

Confinement Boundary

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. The Multi-Purpose Canister (MPC) serves as the Confinement Boundary for contained radioactive materials (Ref. 4.1.13)

Containment Closure

The 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.

For Fermi 2, this condition is met if either Primary Containment or Secondary Containment are functional (i.e. intact).

Emergency Action Level

A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.

Emergency Classification Level

One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual

effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are: Unusual Event (UE), Alert, Site Area Emergency (SAE) and General Emergency (GE).

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 requires Fermi 2 to recommend protective actions for the general public to offsite planning agencies.

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.

Fission Product Barrier Threshold

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

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.

General Emergency

Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE

ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

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 Fermi 2 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 Fermi 2. 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) (Ref. 4.1.8).

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.

Initiating Condition

An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.

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.

Normal Levels

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

Owner Controlled Area

The company property immediately surrounding the PROTECTED AREA security fence. Access is normally limited to people on official business (Ref. 4.1.8).

Projectile

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

Protected Area

An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in UFSAR Figure 1.2-5 Site Plot Plan (Ref. 4.1.10).

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 or nozzle dams).

Refueling Pathway

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

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.

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 Area Emergency

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

Site Boundary

That line beyond which the land is neither owned, nor leased, nor otherwise controlled by DTE Electric. (Ref. 4.1.6).

Unisolable

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

Unplanned

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

Unusual Event

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

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 Acronyms & Abbreviations

°F .... Degrees Fahrenheit
° .... Degrees
‘ .... Feet or Minutes
“ .... Inches or Seconds
ɤ .... Gamma
n .... Neutron
AC .... Alternating Current
AOP .... Abnormal Operating Procedure
APRM .... Average Power Range Monitor
ARM .... Area Radiation Monitor
ATWS .... Anticipated Transient Without Scram
BWR .... Boiling Water Reactor
BWROG .... Boiling Water Reactor Owners Group
CHRRM .... Containment Area High Range Radiation Monitor
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
CS .... Core Spray
CW .... Circulating Water
DBA .... Design Basis Accident
DC .... Direct Current
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
ED .... Emergency Director
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
EPIP .... Emergency Plan Implementing Procedure
EPRI .... Electric Power Research Institute
ERG .... Emergency Response Guideline
ESF .... Engineered Safeguards Feature
FAQ .... Frequently Asked Question
FEMA .... Federal Emergency Management Agency
GE .... General Emergency
HCL .... Heat Capacity Limit
HPCI .... High Pressure Coolant Injection

HSI .... Human System Interface
IC .... Initiating Condition
ID .... Inside Diameter
IPEEE .... Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI .... Independent Spent Fuel Storage Installation
K_{eff} .... Effective Neutron Multiplication Factor
LCO .... Limiting Condition of Operation
LOCA .... Loss of Coolant Accident
LPCI .... Low Pressure Coolant Injection
MCR .... Main Control Room
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
NPP .... Nuclear Power Plant
NRC .... Nuclear Regulatory Commission
NSSS .... Nuclear Steam Supply System
NORAD .... North American Aerospace Defense Command
(NO)UE .... (Notification Of) Unusual Event
NUMARC[1] .... Nuclear Management and Resources Council
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM/ODAM .... Offsite Dose Calculation (Assessment) Manual
ORO .... Off-site Response Organization
PA .... Protected Area
PAG .... Protective Action Guideline
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR .... Pressurized Water Reactor
PS .... Protection System
PSIG .... Pounds per Square Inch Gauge
R .... Roentgen
RB .... Reactor Building
RBSB .... Reactor Building Sub-Basement

1 NUMARC was a predecessor organization of the Nuclear Energy Institute (NEI).

RCC .... Reactor Control Console
RCIC .... Reactor Core Isolation Cooling
RCS .... Reactor Coolant System
Rem, rem, REM .... Roentgen Equivalent Man
RETS .... Radiological Effluent Technical Specifications
RPS .... Reactor Protection System
RPV .... Reactor Pressure Vessel
RSVR .... Reservoir
RW .... Radwaste
RWCU .... Reactor Water Cleanup
SAR .... Safety Analysis Report
SAS .... Safety Automation System
SBO .... Station Blackout
SCBA .... Self-Contained Breathing Apparatus
SGTS .... Standby Gas Treatment System
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
TAF .... Top of Active Fuel
TB .... Turbine Building
TEDE .... Total Effective Dose Equivalent
TSC .... Technical Support Center
UFSAR .... Updated Final Safety Analysis Report

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

This cross-reference is provided to facilitate association and location of a Fermi 2 EAL within the NEI 99-01 IC/EAL identification scheme. Further information regarding the development of the Fermi 2 EALs based on the NEI guidance can be found in the EAL Comparison Matrix.

Fermi 2	NEI 99-01 Rev. 6	
EAL	**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

Fermi 2	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
RG1.3	AG1	3
RG2.1	AG2	1
CU1.1	CU1	1
CU1.2	CU1	2
CU2.1	CU2	1
CU3.1	CU3	1
CU3.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1, 2, 3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA3.1	CA3	1
CA3.2	CA3	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

Fermi 2	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
HU2.1	HU2	1
HU3.1	HU3	1
HU3.2	HU3	2
HU3.3	HU3	3
HU3.4	HU3	4
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

Fermi 2	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
SU6.1	SU5	1
SU6.2	SU5	2
SU7.1	SU6	1, 2, 3
SA1.1	SA1	1
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 Basis

ATTACHMENT 1

EMERGENCY ACTION LEVEL
TECHNICAL BASES

Category R – Abnormal Rad Levels / Rad Effluent

EAL Group: ALL (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 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 or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer.

EAL:

RU1.1 Unusual Event

Reading on **any** Table R-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3)

Note 1: The Emergency Director 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	RB Ventilation	SPING (Ch. 5)	N/A	N/A	N/A	3.3E-3 µCi/cc
Gaseous	SGTS Div. I	SPING (Ch. 7)	N/A	N/A	N/A	4.1E-2 µCi/cc
Gaseous	SGTS Div. I	AXM (Ch. 3)	8.0E+2 µCi/cc	8.0E+1 µCi/cc	8.0E+0 µCi/cc	N/A
Gaseous	SGTS Div. II	SPING (Ch. 7)	N/A	N/A	N/A	4.0E-2 µCi/cc
Gaseous	SGTS Div. II	AXM (Ch. 3)	7.6E+2 µCi/cc	7.6E+1 µCi/cc	7.6E+0 µCi/cc	N/A
Gaseous	RW Ventilation	SPING (Ch. 5)	N/A	N/A	N/A	1.5E-2 µCi/cc
Gaseous	TB Ventilation	SPING (Ch. 5)	N/A	N/A	N/A	2.0E-4 µCi/cc
Liquid	CW RSVR Decant Line	D11-R806	N/A	N/A	1.1E+6 cpm	1.3E+4 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

Plant-Specific

Liquid Releases

Fermi does not perform continuous radioactive liquid releases and no longer performs periodic batch radioactive liquid releases, per administrative controls. However, to provide EALs consistent with the template scheme, a liquid effluent EAL threshold has been developed. (Ref. 2)

Per ODCM Figure 6.0-1, all sources of liquid effluent converge at a common discharge point prior to reaching the environment (Ref. 1).

The D11-K604 Radiation Monitor on the liquid radwaste effluent line provides the alarm and automatic termination of liquid radioactive material releases prior to exceeding 1 MPC at the discharge to Lake Erie. The monitor is located upstream of the Isolation Valve (G11-F733) on the liquid radwaste discharge line and monitors the concentration of liquid effluent before dilution by the circulating water reservoir decant flow (Ref. 2).

The Circulating Water Reservoir (CWR) Decant Line Radiation Monitor (D11-N402) and recorder (D11-R806) provides indication of the concentration of radioactive material in diluted radioactive liquid releases just before discharge to Lake Erie; and thus being the final monitor in the liquid discharge line is the liquid monitor used to address this EAL threshold.

The value shown in Table R-1 column UE represents 2 times the ODCM limit of 1 MPC (Ref. 2).

Gaseous Releases

The column “UE” gaseous release values in Table R-1 represent 2 times the calculated release values associated with the ODCM limit for total body (the skin limit requires a higher release rate value) plus background. (Ref. 1, 2).

For this initiating condition, the applicable effluent monitors are RB SPING, SGTS I SPING, SGTS II SPING, RW SPING and TB SPING (Ref. 2).

Generic

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.

Escalation of the emergency classification level would be via IC RA1.

Fermi Basis Reference(s):

1. Fermi Offsite Dose Calculation Manual
2. EP-EALCALC-FERMI-1401 Radiological Effluent EAL Values Rev. 0
3. 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 ODCM 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 ODCM limits for ≥ 60 min. (Notes 1, 2)	

Note 1: The Emergency Director 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

Basis:

Site Specific

None

Generic

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.

Fermi Basis Reference(s):

1. Fermi Offsite Dose Calculation Manual
2. NEI 99-01 AU1

Category: R – Abnormal Rad Release / 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 thyroid CDE

EAL:

RA1.1 Alert

In the absence of real-time dose assessment, reading on **any** Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4)

Note 1: The Emergency Director 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

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	RB Ventilation	SPING (Ch. 5)	N/A	N/A	N/A	3.3E-3 μCi/cc
Gaseous	SGTS Div. I	SPING (Ch. 7)	N/A	N/A	N/A	4.1E-2 μCi/cc
Gaseous	SGTS Div. I	AXM (Ch. 3)	8.0E+2 μCi/cc	8.0E+1 μCi/cc	8.0E+0 μCi/cc	N/A
Gaseous	SGTS Div. II	SPING (Ch. 7)	N/A	N/A	N/A	4.0E-2 μCi/cc
Gaseous	SGTS Div. II	AXM (Ch. 3)	7.6E+2 μCi/cc	7.6E+1 μCi/cc	7.6E+0 μCi/cc	N/A
Gaseous	RW Ventilation	SPING (Ch. 5)	N/A	N/A	N/A	1.5E-2 μCi/cc
Gaseous	TB Ventilation	SPING (Ch. 5)	N/A	N/A	N/A	2.0E-4 μCi/cc
Liquid	CW RSVR Decant Line	D11-R806	N/A	N/A	1.1E+6 cpm	1.3E+4 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

Plant-Specific

Liquid Releases

The RA1 IC addresses a release of radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.

Per ODCM Figure 6.0-1, all sources of liquid effluent converge at a common discharge point prior to reaching the environment (Ref. 1).

The D11-K604 Radiation Monitor on the liquid radwaste effluent line provides the alarm and automatic termination of liquid radioactive material releases prior to exceeding 1 MPC at the discharge to Lake Erie. The monitor is located upstream of the Isolation Valve (G11-F733) on the liquid radwaste discharge line and monitors the concentration of liquid effluent before dilution by the circulating water reservoir decant flow (Ref. 2).

The Circulating Water Reservoir (CWR) Decant Line Radiation Monitor (D11-N402) and recorder (D11-R806) provides indication of the concentration of radioactive material in diluted radioactive liquid releases just before discharge to Lake Erie; and thus being the final monitor in the liquid discharge line is the liquid monitor used to address this EAL threshold (Ref. 1, 2).

The value shown in Table R-1 column Alert for liquid releases represents 10 mRem for one hour of exposure (Ref. 2).

Gaseous Releases

For gaseous releases, the preferred method for classification is by means of the computerized dose assessment program incorporating actual meteorology. This method is preferred since it eliminates uncertainty associated with assumed meteorology and source term data.

For this initiating condition, the applicable gaseous effluent monitors are the Division I and II AXMs.

The column “ALERT” gaseous release values in Table R-1 represent offsite dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 10 mRem TEDE or 50 mRem thyroid CDE (1% of the EPA PAGs) (Ref. 2).

Generic

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.

Fermi Basis Reference(s):

1. Fermi Offsite Dose Calculation Manual
2. EP-EALCALC-FERMI-1401 Radiological Effluent EAL Values Rev. 0
3. NEI 99-01 AA1

Category: R – Abnormal Rad Release / 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 thyroid CDE

EAL:

RA1.2	**Alert**
Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE 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):

SITE BOUNDARY - That line beyond which the land is neither owned, nor leased, nor otherwise controlled by DTE Electric.

Basis:

<u>Plant-Specific</u>

Calculated dose from airborne sources using computerized dose assessment model incorporating current meteorology indicates greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY (Ref. 1).

<u>Generic</u>

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.

Fermi Reference(s):

1. EP-542 Computer-Based Offsite Dose Assessment - Airborne Release, Rev. 11
2. NEI 99-01 AA1

Category: R – Abnormal Rad Release / 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 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 thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure (Notes 1, 2)

Note 1: The Emergency Director 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):

SITE BOUNDARY - That line beyond which the land is neither owned, nor leased, nor otherwise controlled by DTE Electric.

Basis:

Plant-Specific

Dose assessments based on liquid releases are manual calculations performed per Offsite Dose Calculation Manual (Ref. 1).

Generic

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.

Fermi Reference(s):

1. Fermi 2 Offsite Dose Calculation Manual
2. NEI 99-01 AA1

Category: R – Abnormal Rad Release / 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 thyroid CDE

EAL:

RA1.4 Alert

Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY:

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

(Notes 1, 2)

Note 1: The Emergency Director 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):

SITE BOUNDARY - That line beyond which the land is neither owned, nor leased, nor otherwise controlled by DTE Electric.

Basis:

Plant-Specific

EP-220, Personnel Monitoring and Radiological Emergency Teams, provides guidance for emergency or post-accident radiological environmental monitoring (Ref. 1).

Generic

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.

Fermi Reference(s):

1. EP-220, Personnel Monitoring and Radiological Emergency Teams, Rev. 19
2. NEI 99-01 AA1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.1 Site Area Emergency

In the absence of real-time dose assessment, reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min.
(Notes 1, 2, 3, 4)

Note 1: The Emergency Director 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

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	RB Ventilation	SPING (Ch. 5)	N/A	N/A	N/A	3.3E-3 μCi/cc
Gaseous	SGTS Div. I	SPING (Ch. 7)	N/A	N/A	N/A	4.1E-2 μCi/cc
Gaseous	SGTS Div. I	AXM (Ch. 3)	8.0E+2 μCi/cc	8.0E+1 μCi/cc	8.0E+0 μCi/cc	N/A
Gaseous	SGTS Div. II	SPING (Ch. 7)	N/A	N/A	N/A	4.0E-2 μCi/cc
Gaseous	SGTS Div. II	AXM (Ch. 3)	7.6E+2 μCi/cc	7.6E+1 μCi/cc	7.6E+0 μCi/cc	N/A
Gaseous	RW Ventilation	SPING (Ch. 5)	N/A	N/A	N/A	1.5E-2 μCi/cc
Gaseous	TB Ventilation	SPING (Ch. 5)	N/A	N/A	N/A	2.0E-4 μCi/cc
Liquid	CW RSVR Decant Line	D11-R806	N/A	N/A	1.1E+6 cpm	1.3E+4 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

Plant-Specific

For gaseous releases, the preferred method for classification is by means of the computerized dose assessment program incorporating actual meteorology and effluent monitor readings. This method is preferred since it eliminates uncertainty associated with assumed meteorology and source term data.

For this initiating condition, the applicable gaseous effluent monitors are the Division I and II AXMs.

The column “SAE” gaseous release values in Table R-1 represent offsite dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 100 mRem TEDE or 500 mRem thyroid CDE (10% of the EPA PAGs) (Ref. 1).

Generic

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.

Fermi Basis Reference(s):

1. EP-EALCALC-FERMI-1401 Radiological Effluent EAL Values Rev. 0
2. NEI 99-01 AS1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.2	**Site Area Emergency**
Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (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):

SITE BOUNDARY - That line beyond which the land is neither owned, nor leased, nor otherwise controlled by DTE Electric.

Basis:

Plant-Specific

Calculated dose from airborne sources using computerized dose assessment model incorporating current meteorology indicates greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (Ref. 1).

Generic

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.

Fermi Reference(s):

1. EP-542 Computer-Based Offsite Dose Assessment - Airborne Release, Rev. 11
2. NEI 99-01 AS1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.3 Site Area Emergency

Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY:

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

(Notes 1, 2)

Note 1: The Emergency Director 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):

SITE BOUNDARY - That line beyond which the land is neither owned, nor leased, nor otherwise controlled by DTE Electric.

Basis:

Plant-Specific

EP-220, Personnel Monitoring and Radiological Emergency Teams, provides guidance for emergency or post-accident radiological environmental monitoring (Ref. 1).

Generic

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.

Fermi Reference(s):

1. EP-220, Personnel Monitoring and Radiological Emergency Teams, Rev. 19
2. NEI 99-01 AS1

Category: R – Abnormal Rad Release / 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 thyroid CDE

EAL:

RG1.1 General Emergency

In the absence of real-time dose assessment, reading on **any** Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min.
(Notes 1, 2, 3, 4)

Note 1: The Emergency Director 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

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	RB Ventilation	SPING (Ch. 5)	N/A	N/A	N/A	3.3E-3 µCi/cc
	SGTS Div. I	SPING (Ch. 7)	N/A	N/A	N/A	4.1E-2 µCi/cc
		AXM (Ch. 3)	8.0E+2 µCi/cc	8.0E+1 µCi/cc	8.0E+0 µCi/cc	N/A
	SGTS Div. II	SPING (Ch. 7)	N/A	N/A	N/A	4.0E-2 µCi/cc
		AXM (Ch. 3)	7.6E+2 µCi/cc	7.6E+1 µCi/cc	7.6E+0 µCi/cc	N/A
	RW Ventilation	SPING (Ch. 5)	N/A	N/A	N/A	1.5E-2 µCi/cc
	TB Ventilation	SPING (Ch. 5)	N/A	N/A	N/A	2.0E-4 µCi/cc
Liquid	CW RSVR Decant Line	D11-R806	N/A	N/A	1.1E+6 cpm	1.3E+4 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

Plant-Specific

For gaseous releases, the preferred method for classification is by means of the computerized dose assessment program incorporating actual meteorology and effluent monitor readings. This method is preferred since it eliminates uncertainty associated with assumed meteorology and source term data.

For this initiating condition, the applicable gaseous effluent monitors are the Division I and II AXMs.

The column “GE” gaseous release values in Table R-1 represent offsite dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 1000 mRem TEDE or 5000 mRem thyroid CDE (100% of the EPA PAGs) (Ref. 1).

Generic

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.

Fermi Basis Reference(s):

1. EP-EALCALC-FERMI-1401 Radiological Effluent EAL Values Rev. 0
2. NEI 99-01 AG1

Category: R – Abnormal Rad Release / 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 thyroid CDE

EAL:

RG1.2	**General Emergency**
Dose assessment using actual meteorology indicates doses > 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond the SITE BOUNDARY (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):

SITE BOUNDARY - That line beyond which the land is neither owned, nor leased, nor otherwise controlled by DTE Electric.

Basis:

Plant-Specific

Calculated dose from airborne sources using computerized dose assessment model incorporating current meteorology indicates greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE (Ref. 1).

Generic

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.

Fermi Reference(s):

1. EP-542 Computer-Based Offsite Dose Assessment - Airborne Release, Rev. 11
2. NEI 99-01 AG1

Category: R – Abnormal Rad Release / 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 thyroid CDE

EAL:

RG1.3 General Emergency

Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY:

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

(Notes 1, 2)

Note 1: The Emergency Director 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):

SITE BOUNDARY - That line beyond which the land is neither owned, nor leased, nor otherwise controlled by DTE Electric.

Basis:

Plant-Specific

EP-220, Personnel Monitoring and Radiological Emergency Teams, provides guidance for emergency or post-accident radiological environmental monitoring (Ref. 1).

Generic

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.

Fermi Reference(s):

1. EP-220, Personnel Monitoring and Radiological Emergency Teams, Rev. 19
2. NEI 99-01 AG1

Category: R – Abnormal Rad Release / 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:

- 2D1, FUEL POOL WATER LEVEL LOW alarm
- Floodup Level Transmitter (when in service)
- Visual observation

AND

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

- RB5 Spent Fuel Pool ARM (Ch. 15)
- RB5 Refuel Floor Lo Range ARM (Ch. 17)
- RB5 Refuel Floor Hi Range ARM (Ch. 18)

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 (cattle chute) comprise the refueling pathway.

Basis:

Plant-Specific

Indications of decreasing level include:

- Alarm 2D1, FUEL POOL WATER LEVEL LOW (Ref. 1, 2)
- Floodup Level Transmitter (when in service)

During refueling operations with the fuel pool gates removed, the RPV floodup level instrumentation (B21-N027) and the Rx Vessel Core Plate dp transmitter (B21-N032) are capable of displaying the common level of the reactor cavity and the spent fuel pool. There is a Low Reactor Vessel/Fuel Pool Water Level Alarm that can be connected to these transmitters as well as a 5th Floor Alarm Unit that can be used to warn of the loss of shielding. (Ref. 3)

- Visual observation of reactor cavity and spent fuel pool level from the Refueling Floor

Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment. Technical Specification LCO 3.7.7 requires at least 22 ft of water above irradiated fuel assemblies seated in the spent fuel pool storage racks. Technical Specification LCO 3.9.6 requires at least 20 ft 6 in. of water above the top of the reactor vessel flange in the refueling cavity during refueling operations. This maintains sufficient water level in the fuel transfer canal, refueling cavity, and spent fuel pool to retain iodine fission product activity in the water in the event of a fuel handling accident (Ref. 4, 5). The spent fuel pool low level alarm is actuated by level switch G41-N001B four inches below normal level. (Ref. 2)

Radiation monitors that may indicate a loss of shielding above irradiated fuel include (Ref. 6):

- RB5 Spent Fuel Pool ARM (Ch. 15)
- RB5 Fuel Floor Lo Range ARM (Ch. 17)
- RB5 Fuel Floor Hi Range ARM (Ch. 18)

Generic

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.

Fermi Reference(s):

1. AOP 20.708 Loss of FPCCU
2. ARP 2D1 Fuel Pool Water Level Low
3. SOP 23.800.06 Rev. 10 Reactor Vessel Water Level Monitoring During Refueling Operations
4. Technical Specifications LCO 3.7.7 Spent Fuel Pool water Level
5. Technical Specifications LCO 3.9.6 RPV Water Level
6. ARP 16D1 RB REFUELING FIFTH FLOOR HIGH RADN
7. NEI 99-01 AU2

Category: R – Abnormal Rad Release / 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 (cattle chute) comprise the refueling pathway.

Basis:

Plant-Specific

Indications of decreasing water level with the potential to uncover irradiated fuel include:

- Floodup Level Transmitter (when in service)

  During refueling operations with the fuel pool gates removed, the RPV floodup level instrumentation (B21-N027) and the Rx Vessel Core Plate dp transmitter (B21-N032) are capable of displaying the common level of the reactor cavity and the spent fuel pool. There's a Low Reactor Vessel/Fuel Pool Water Level Alarm that can be connected to these transmitters as well as a 5th Floor Alarm Unit that can be used to warn of the loss of shielding. (Ref. 1)

- Visual observation of reactor cavity and/or spent fuel pool level from the Refueling Floor

Technical Specification LCO 3.7.7 requires at least 22 ft of water above irradiated fuel assemblies seated in the spent fuel pool storage racks. Technical Specification LCO 3.9.6 requires at least 20 ft 6 in. of water above the top of the reactor vessel flange in the refueling cavity during refueling operations. This maintains sufficient water level in the fuel

transfer canal, refueling cavity, and spent fuel pool to retain iodine fission product activity in the water in the event of a fuel handling accident. (Ref. 2, 3) Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment.

<u>Generic</u>

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 applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with EU1.1.

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.

Fermi Reference(s):

1. SOP 23.800.06 Rev. 10 Reactor Vessel Water Level Monitoring During Refueling Operations
2. Technical Specifications LCO 3.7.7 Spent Fuel Pool water Level
3. Technical Specifications LCO 3.9.6 RPV Water Level
4. NEI 99-01 AA2

Category: R – Abnormal Rad Release / 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:

- RB5 Refuel Floor Hi Range ARM (Ch. 18) alarm
- RBHVAC Vent Exhaust Radiation Monitor $\geq$ 16,000 cpm
- Fuel Pool Vent Exhaust Radiation Monitor $\geq$ 5 mR/hr

Mode Applicability:

All

Definition(s):

None

Basis:

Plant-Specific

When considering escalation, information may come from:

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

Radiation monitors and associated indications specified in this EAL are:

- RB5 Fuel Floor Hi Range ARM (Ch. 18) alarm:
  This high range area radiation monitor alarms at 1,000 mr/hr and provides confirming indication of possible damage to irradiated fuel (Ref. 1, 2, 3)

- RBHVAC Vent Exhaust Radiation Monitor $\geq$ 16,000 cpm:
  This monitor provides indication of the release of radioactive fission products to the Reactor Building atmosphere as a result of damaged irradiated fuel. A reading of $\geq$ 16,000 cpm requires entry into the Secondary Containment Control EOP (Ref. 4).
- Fuel Pool Vent Exhaust Radiation Monitor $\geq$ 5 mR/hr:
  This monitor also provides indication of the release of radioactive fission products to the Refueling Floor atmosphere (Reactor Building) as a result of damaged irradiated fuel. A reading of $\geq$ 5 mR/hr requires entry into the Secondary Containment Control EOP (Ref. 4).

Plant procedures require termination of fuel and core component movements and evacuation of the Reactor Building if elevated radiation levels are detected. All core alternations are stopped and transient fuel assemblies and core components are placed in a safe position in the reactor vessel, Spent Fuel Pool or fuel transfer canal to the extent practicable (Ref. 2, 3).

Generic

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.

Fermi Reference(s):

1. ARP 16D1 RB REFUELING FIFTH FLOOR HIGH RADN
2. AOP 20.710.01 Refueling Floor High Radiation
3 AOP 20.000.02 Abnormal Release of Radioactive Material
4. EOP 29.100.01 SH5 Secondary Containment and Rad Release
5. NEI 99-01 AA2

Category: R – Abnormal Rad Release / 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 Level 2 as indicated by level $\leq$ 33 ft. on G41R601A/B.	

Mode Applicability:

All

Definition(s):

None

Basis:

Plant-Specific

The Fermi SFP is located on the Reactor Building 5th floor. The surface of the water is normally maintained at plant elevation 683.5 ft. (Level 1) by scuppers that act as skimmers and wave suppressors. This results in a minimum water depth of 7 ft. above the top of the fuel while it is being moved above storage racks. Pool water level indication is painted on the north and east walls of the pool starting at 18 ft. above the stored fuel assemblies (Ref. 1).

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level at which radiation level would still allow personnel access near the pool, 18 ft. above the top of the fuel racks (Level 2 or ele. 679 ft. 1/8 in.) and SFP level at the top of the fuel racks (Level 3 or ele. 661 ft. 1/8 in.). Remote SFP level indication is available in the control room on level indicator G41R601A Panel H11P601. An indicated level of 33 ft. corresponds to the Level 2 setpoint (Ref. 2).

Generic

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.

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.

Fermi Reference(s):

1. UFSAR Section 9.1.2.2.1
2. Engineering Design Package (EDP) 37088
3. NEI 99-01 AA2

Category: R – Abnormal Rad Release / 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 rate > 15 mR/hr in EITHER of the following areas:

- AB3 Control Room (ARM Channel 6)
- Central Alarm Station (by survey)

Mode Applicability:

All

Definition(s):

None

Basis:

Plant-Specific

ARM Channel 6 (D21-N106) is the permanently installed Control Room area radiation monitor and, along with local radiation surveys, may be used to assess this EAL threshold (Ref. 1). Permanently installed area radiation monitoring is not installed in the CAS and, therefore, radiation levels in this area must be assessed with local radiation survey techniques

Generic

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

Fermi Basis Reference(s):

1. SOP 23.611 Area Radiation Monitoring System
2. NEI 99-01 AA3

Category: R – Abnormal Rad Release / 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 Shutdown Rooms/Areas

Room/Area	Mode Applicability
• Relay Room	All
• AB3-DC MCC Area	Mode 3
• RB1-F17	Mode 3
• RB1-F11	Mode 3

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.

Basis:

Plant-Specific

The rooms/areas and associated mode applicability specified in Table R-2 are those that contain equipment which require a manual/local action as specified in operating procedures used for normal operation, cooldown and shutdown. This table excludes rooms/areas that may have procedurally directed actions that are of an administrative nature (normal rounds or routine inspections) or that are not crucial to the conduct of safe operation, cooldown and shutdown.

Specifically:

- Control Room & Relay Room in all modes (Control Room is not included as it is addressed in RA3.1)
- AB3-DC MCC Area – Access is required when in Mode 3 to install power fuses and close the MCC for E1150-F008 which must be performed to align shutdown cooling suction path.
- RB1-F17 – Access is required when in Mode 3 to place the permissive switch in OPERATE for E11F610A if Div 1 RHR is being placed in shutdown cooling. This step must be performed to warmup the shutdown cooling piping.
- RB1-F11 – Access is required when in Mode 3 to place the permissive switch in OPERATE for E11F610B if Div 2 RHR is being placed in shutdown cooling. This step must be performed to warmup the shutdown cooling piping.

(Ref. 1).

<u>Generic</u>

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

Fermi Basis Reference(s):

1. Operating Procedures (normal plant operations, cooldown or shutdown), manual / local actions:
   a. 22.000.03 - Power Operation 25% to 100% to 25%
   b. 22.000.04 - Plant Shutdown From 25% Power
   c. 22.000.05 - Pressure/Temp Monitoring During Heatup and Cooldown
   d. 23.202 - High Pressure Coolant Injection System
   e. 23.205 - Residual Heat Removal System
   f. 23.206 - Reactor Core Isolation Cooling System
   g. 23.427 - Primary Containment Isolation System
   h. 23.610 - Reactor Protection System (RPS)
   i. MGA03 - Procedure Use and Adherence
2. NEI 99-01 AA3

Category: R – Abnormal Rad Release / 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 Level 3 as indicated by level $\leq$ 0 ft. 3 in. on G41R601A/B.	

Mode Applicability:

All

Definition(s):

None

Basis:

Plant-Specific

The Fermi SFP is located on the Reactor Building 5th floor. The surface of the water is normally maintained at plant elevation 683.5 ft. (Level 1) by scuppers that act as skimmers and wave suppressors. This results in a minimum water depth of 7 ft. above the top of the fuel while it is being moved above storage racks. Pool water level indication is painted on the north and east walls of the pool starting at 18 ft. above the stored fuel assemblies (Ref. 1).

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level at which radiation level would still allow personnel access near the pool, 18 ft. above the top of the fuel racks (Level 2 or ele. 679 ft. 1/8 in.) and SFP level at the top of the fuel racks (Level 3 or ele. 661 ft. 1/8 in.). Remote SFP level indication is available in the control room on level indicator G41R601A Panel H11P601. An indicated level of 0 ft. 3 in. corresponds to the Level 3 setpoint (Ref. 2).

Generic

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.

Fermi Reference(s):

1. UFSAR Section 9.1.2.2.1
2. Engineering Design Package (EDP) 37088
3. NEI 99-01 AS2

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RG2.1	**General Emergency**
Spent fuel pool level cannot be restored to at least Level 3 as indicated by level > 0 ft. 3 in. on G41R601A/B for $\geq$ 60 min. (Note 1)	

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

Mode Applicability:

All

Definition(s):

None

Basis:

Plant-Specific

The Fermi SFP is located on the Reactor Building 5th floor. The surface of the water is normally maintained at plant elevation 683.5 ft. (Level 1) by scuppers that act as skimmers and wave suppressors. This results in a minimum water depth of 7 ft. above the top of the fuel while it is being moved above storage racks. Pool water level indication is painted on the north and east walls of the pool starting at 18 ft. above the stored fuel assemblies (Ref. 1).

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level at which radiation level would still allow personnel access near the pool, 18 ft. above the top of the fuel racks (Level 2 or ele. 679 ft. 1/8 in.) and SFP level at the top of the fuel racks (Level 3 or ele. 661 ft. 1/8 in.). Remote SFP level indication is available in the control room on level indicator G41R601A Panel H11P601. An indicated level of 0 ft. 3 in. corresponds to the Level 3 setpoint (Ref. 2).

Generic

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.

Fermi Reference(s):

1. UFSAR Section 9.1.2.2.1
2. Engineering Design Package (EDP) 37088
3. NEI 99-01 AG2

Category C – Cold Shutdown / Refueling System Malfunction

EAL Group: Cold Conditions (RCS temperature ≤ 200ºF); 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 4160 VAC essential buses 64B/64C and 65E/65F.

3. RCS Temperature

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

4. 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 130 VDC ESF buses.

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: UNPLANNED loss of RPV inventory for 15 minutes or longer.

EAL:

CU1.1	**Unusual Event**
UNPLANNED loss of reactor coolant results in RPV water level below the established control band for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director 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.

Basis:

<u>Plant-Specific</u>

With the plant in Cold Shutdown, RPV water level is required to be maintained above 214 in. and normally maintained in a level band of 220 to 255 in. above TAF (Ref. 1). However, if RPV level is being controlled below the normal band, 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 Specification LCO 3.9.6 requires at least 20 ft 6 in. of water above the top of the reactor vessel flange in the refueling cavity during refueling operations) (Ref. 4). The reactor vessel flange mating surface is 379 in. above TAF (Ref. 2). RPV level can be monitored by one or more of the following (Ref. 3):

- Flood-up Level indicator B21-R605 (+160 in. to +560 in.)

- Wide Range Level indicators B21-R604A/B (+10 in. to +220 in.)
- Narrow Range Level indicators C32-R606A/B/C/D (+160 in. to +220 in.)
- Floodup Level Transmitter (when in service)

  During refueling operations with the fuel pool gates removed, the RPV floodup level instrumentation (B21-N027) and the Rx Vessel Core Plate dp transmitter (B21-N032) are capable of displaying the common level of the reactor cavity and the spent fuel pool. (Ref. 2)
- Visual observation of reactor cavity level from the Refueling Floor or by remote video display (when available)

Regardless of where RPV level is intentionally being controlled, either above or below the reactor vessel flange, as in Cold Shutdown, 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.

<u>Generic</u>

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band). This condition 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.

Fermi Basis Reference(s):

1. GOP 22.000.04 Plant Shutdown From 25% Power
2. SOP 23.800.06 Reactor Vessel Water Level Monitoring During Refueling Operations
3. UFSAR Section 7 Instrumentation and Controls Table 7.5-1 Control Room Level Indication
4. Technical Specification LCO 3.9.6 RPV Water Level
5. NEI 99-01 CU1

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 Floor Drain Sump • Drywell Equipment Drain Sump • RB Floor Drain Sumps • RB Equipment Drain Sumps • Torus • 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.

Basis:

Plant-Specific

RPV level can be monitored by one or more of the following (Ref. 1):

- Flood-up Level indicator B21-R605 (+160 in. to +560 in.)
- Wide Range Level indicators B21-R604A/B (+10 in. to +220 in.)
- Narrow Range Level indicators C32-R606A/B/C/D (+160 in. to +220 in.)

- Floodup Level Transmitter (when in service)

  During refueling operations with the fuel pool gates removed, the RPV floodup level instrumentation (B21-N027) and the Rx Vessel Core Plate dp transmitter (B21-N032) are capable of displaying the common level of the reactor cavity and the spent fuel pool. (Ref. 3)

- Visual observation of reactor cavity level from the Refueling Floor

In this EAL, all water level indication is unavailable, and the RCS inventory loss must be detected by sump or tank level changes (Table C-1). Plant design and procedures provide the capability to detect and assess RCS leakage (Ref. 2).

<u>Generic</u>

This IC addresses a loss of the ability to monitor RPV level concurrent with indications of coolant leakage. This condition 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 RCS.

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

Fermi Basis Reference(s):

1. UFSAR Section 7 Instrumentation and Controls Table 7.5-1 Control Room Level Indication
2. UFSAR Section 5.2.7 Reactor Coolant Pressure Boundary Leak Detection System

3. SOP 23.800.06 Rev. 10 Reactor Vessel Water Level Monitoring During Refueling Operations
4. NEI 99-01 CU1

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 water level < 111 in. above TAF (Level 2)	

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

None

Basis:

Plant-Specific

When reactor vessel water level drops to 110.8 in. (rounded to 111in.) above TAF high pressure steam-driven injection sources HPCI (ECCS) and RCIC actuate (Ref. 1). Although these systems cannot restore RCS inventory in the cold condition, the Low-Low (Level 2) ECCS actuation setpoint is operationally significant and is indicative of a loss of RCS inventory significantly below the normally established control band specified in CU1.1.

Generic

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 111 in above TAF 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 Decay Heat Removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

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

Fermi Basis Reference(s):

1. TRM Table TR3.3.5.1-1 Emergency Core Cooling System Instrumentation
2. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Loss of RPV inventory

EAL:

CA1.2 Alert

RPV water 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

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

Table C-1 Sumps & Tanks
• Drywell Floor Drain Sump • Drywell Equipment Drain Sump • RB Floor Drain Sumps • RB Equipment Drain Sumps • Torus • 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.

Basis:

Plant-Specific

RPV level can be monitored by one or more of the following (Ref. 1):

- Flood-up Level indicator B21-R605 (+160 in. to +560 in.)

- Wide Range Level indicators B21-R604A/B (+10 in. to +220 in.)
- Narrow Range Level indicators C32-R606A/B/C/D (+160 in. to +220 in.)
- Floodup Level Transmitter (when in service)

  During refueling operations with the fuel pool gates removed, the RPV floodup level instrumentation (B21-N027) and the Rx Vessel Core Plate dp transmitter (B21-N032) are capable of displaying the common level of the reactor cavity and the spent fuel pool. (Ref. 3)

- Visual observation of reactor cavity level from the Refueling Floor

In this EAL, all water level indication is unavailable for greater then 15 minutes, and the RCS inventory loss must be detected by sump or tank level changes (Table C-1). Plant design and procedures provide the capability to detect and assess RCS leakage (Ref. 2).

Generic

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 RCS.

The 15-minute duration for the loss of level indication was chosen because it is half of the EAL duration specified in IC CS1.

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

Fermi Basis Reference(s):

1. UFSAR Section 7 Instrumentation and Controls Table 7.5-1 Control Room Level Indication

2. UFSAR Section 5.2.7 Reactor Coolant Pressure Boundary Leak Detection System
3. SOP 23.800.06 Rev. 10 Reactor Vessel Water Level Monitoring During Refueling Operations
4. NEI 99-01 CA1

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 water level < 32 in. above TAF (Level 1)

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The 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.

For Fermi 2, this condition is met if either Primary Containment or Secondary Containment are functional (i.e. intact).

Basis:

Plant-Specific

When reactor vessel water level drops to 31.8 in. (rounded to 32 in.) above TAF low pressure ECCS such as LPCI and Core Spray actuate (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.

<u>Generic</u>

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

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in 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.

Fermi Basis Reference(s):

1. TRM Table TR3.3.5.1-1 Emergency Core Cooling System Instrumentation
2. 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:

CS1.2 Site Area Emergency
CONTAINMENT CLOSURE established
AND
RPV water level < 0 in.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The 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.

For Fermi 2, this condition is met if either Primary Containment or Secondary Containment are functional (i.e. intact).

Basis:

Plant-Specific

When RPV water level drops to 0 in. (TAF) core uncovery is about to occur (Ref. 1).

Generic

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.

Fermi Basis Reference(s):

1. EOP Support Documentation Section 1 Plant Specific Technical Guideline
2. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.3 Site Area Emergency

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

AND

Core uncovery is indicated by **EITHER** of the following:

- RB5 Refuel Floor Hi Range ARM (Ch. 18) alarm
- UNPLANNED increase in **any** Table C-1 sump or tank levels due to a loss of RPV inventory

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

Table C-1 Sumps & Tanks
• Drywell Floor Drain Sump • Drywell Equipment Drain Sump • RB Floor Drain Sumps • RB Equipment Drain Sumps • Torus • 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.

Basis:

Plant-Specific

If all means of level monitoring are not available, the RCS inventory loss may be detected by the Fuel Floor area radiation monitors or indication or sump/tank level increases:

- In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed area radiation monitors. RB5 Fuel Floor Hi Range ARM (Ch. 18) is located on the Refuel Floor in the Reactor Building and is designed to provide monitoring of radiation due to a fuel handling event or loss of shielding during refueling operations. If this radiation monitor reaches and exceeds the alarm setpoint of 1,000 mr/hr, a loss of inventory with potential to uncover the core is likely to have occurred. D11-N443A/B are the Containment High Range Radiation Monitors but they are not located in the Containment with sufficient line-of-sight to the irradiated fuel in the reactor vessel to be of use in detecting loss of inventory above the core (Ref. 1, 2).
- If water level monitoring capability is unavailable, the reactor vessel inventory loss may be detected by sump or tank level changes (Table C-1). Plant design and procedures provide the capability to detect and assess RCS leakage (Ref. 3).

Generic

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

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in 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.

Fermi Basis Reference(s):

1. ARP 16D1 RB REFUELING FIFTH FLOOR HIGH RADN
2. AOP 20.710.01 Refueling Floor High Radiation
3. UFSAR Section 5.2.7 Reactor Coolant Pressure Boundary Leak Detection System
4. NEI 99-01 CS1

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 water level < 0 in. for ≥ 30 min. (Note 1)

AND

Any of the following indications of containment challenge:

- CONTAINMENT CLOSURE **not** established (Note 6)
- Primary Containment hydrogen concentration > 6%
- UNPLANNED increase in Primary Containment pressure
- Exceeding Secondary Containment Control **Max Safe** Operating Area Radiation Level on channel 14 ARM (RBSB Torus Room)

Note 1: The Emergency Director 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.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The 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.

For Fermi 2, this condition is met if either Primary Containment or Secondary Containment are functional (i.e. intact).

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.

Basis:

Plant-Specific

When RPV water level drops to 0 in. (TAF) core uncovery is about to occur (Ref. 1).

Four indications are associated with a challenge to Containment:

- 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. 1). In Cold Shutdown and Refueling modes it is assumed that the Primary Containment is de-inerted.
- An unplanned pressurization that can breach the containment barrier signifies a challenge to the Primary Containment pressure retaining capability which is dependent on the status of either containment integrity or CONTAINMENT CLOSURE. If containment integrity is established for full power operation, a breach could occur if the design Primary Containment pressure is exceeded (62 psig) (Ref. 2). For this condition, a small unplanned pressure rise above atmospheric pressure does not challenge containment. If in refueling operations, however, a breach could occur if the unplanned pressure rise exceeded the capability of a temporary containment seal used to establish CONTAINMENT CLOSURE. This would occur at a much lower pressure than the containment design pressure.
- The use of secondary containment radiation monitors provides indication of increased release that may be indicative of a challenge to Primary Containment. The Secondary Containment Control EOP Max Safe area radiation values have been selected because these values are easily recognizable and have a defined basis. (Ref. 1, 3)

The only Secondary Containment Maximum Safe Operating Radiation Level that can be directly obtained in the Control Room is the RBSB Torus Room on ARM Channel 14. Other installed area radiation monitors listed in the EOP **Max Safe** table are NOT used in this determination because of instrument range limitations that would require a survey to determine the EOP referenced **Max Safe** value. Surveys initiated for EOP **Max Safe** assessments do not generally provide timely information to determine classification of this event.

<u>Generic</u>

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*.

Fermi Basis Reference(s):

1. EOP Support Documentation Section 1 Plant Specific Technical Guideline
2. UFSAR Table 6.2-1 Containment Parameters
3. EOP 29.100.01 Sheet 5 Secondary Containment and Rad Release
4. 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 water level cannot be monitored for ≥ 30 min. (Note 1)

AND

Core uncovery is indicated by **EITHER** of the following:

- RB5 Refuel Floor Hi Range ARM (Ch. 18) alarm
- UNPLANNED increase in **any** Table C-1 sump or tank levels due to a loss of RPV inventory

AND

Any of the following indications of containment challenge:

- CONTAINMENT CLOSURE not established (Note 6)
- Primary Containment hydrogen concentration > 6%
- UNPLANNED increase in Primary Containment pressure
- Exceeding Secondary Containment Control **Max Safe** Operating Area Radiation Level on channel 14 ARM (RBSB Torus Room)

Note 1: The Emergency Director 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 Floor Drain Sump • Drywell Equipment Drain Sump • RB Floor Drain Sumps • RB Equipment Drain Sumps • Torus • Visual Observation

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The 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.

For Fermi 2, this condition is met if either Primary Containment or Secondary Containment are functional (i.e. intact).

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.

Basis:

Plant-Specific

If all means of level monitoring are not available, the RCS inventory loss may be detected by the Fuel Floor area radiation monitors or indication or sump/tank level increases:

- In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed area radiation monitors. RB5 Fuel Floor Hi Range ARM (Ch. 18) is located on the Refuel Floor in the Reactor Building and is designed to provide monitoring of radiation due to a fuel handling event or loss of shielding during refueling operations. If this radiation monitor reaches and exceeds the alarm setpoint of 1,000 mr/hr, a loss of inventory with potential to uncover the core is likely to have occurred. D11-N443A/B are the Containment High Range Radiation Monitors but they are not located in the Containment with sufficient line-of-sight to the irradiated fuel in the reactor vessel to be of use in detecting loss of inventory above the core (Ref. 1, 2).
- If water level monitoring capability is unavailable, the reactor vessel inventory loss may be detected by sump or tank level changes (Table C-1). Plant design and procedures provide the capability to detect and assess RCS leakage (Ref. 3).

Four indications are associated with a challenge to Containment:

- 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. 4). In Cold Shutdown and Refueling modes it is assumed that the Primary Containment is de-inerted.
- An unplanned pressurization that can breach the containment barrier signifies a challenge to the Primary Containment pressure retaining capability which is dependent on the status of either containment integrity or CONTAINMENT CLOSURE. If containment integrity is established for full power operation, a breach could occur if the design Primary Containment pressure is exceeded (62 psig) (Ref. 5). For this condition, a small unplanned pressure rise above atmospheric pressure does not challenge containment. If in refueling operations, however, a breach could occur if the unplanned pressure rise exceeded the capability of a temporary containment seal used to establish CONTAINMENT CLOSURE. This would occur at a much lower pressure than the containment design pressure.
- The use of secondary containment radiation monitors provides indication of increased release that may be indicative of a challenge to Primary Containment. The Secondary Containment Control EOP Max Safe area radiation values have been selected because these values are easily recognizable and have a defined basis. (Ref. 4, 6)

The only Secondary Containment Maximum Safe Operating Radiation Level that can be directly obtained in the Control Room is the RBSB Torus Room on ARM Channel 14. Other installed area radiation monitors listed in the EOP **Max Safe** table are NOT used in

this determination because of instrument range limitations that would require a survey to determine the EOP referenced **Max Safe** value. Surveys initiated for EOP **Max Safe** assessments do not generally provide timely information to determine classification of this event.

<u>Generic</u>

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*.

Fermi Basis Reference(s):

1. ARP 16D1 RB REFUELING FIFTH FLOOR HIGH RADN
2. AOP 20.710.01 Refueling Floor High Radiation
3. UFSAR Section 5.2.7 Reactor Coolant Pressure Boundary Leak Detection System
4. EOP Support Documentation Section 1 Plant Specific Technical Guideline
5. UFSAR Section 6.2.1.2.1 Primary Containment
6. EOP 29.100.01 Sheet 5 Secondary Containment and Rad Release
7. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Essential 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 4160V essential Division I (64B/64C) and Division II (65E/65F) reduced to a single power source (Table C-2) for ≥ 15 min. (Note 1, 10)

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

Note 10: Credit may be taken for one of the four CTGs as an onsite AC power source only if the CTG is already aligned and capable of powering an essential bus within 15 min.

Table C-2 AC Power Sources
Offsite: • System Service Transformer 64 (Div I) • System Service Transformer 65 (Div II) **Onsite:** • EDGs 11/12 (Div I) • EDGs 13/14 (Div II) • CTG 11-1, 11-2, 11-3 or 11-4

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.

Basis:

Plant-Specific

Table C-2 lists AC sources capable of powering essential AC buses. For emergency classification purposes, “capability” means that an AC power source is available to the essential divisional buses, whether or not the buses are currently powered from it.

This EAL is indicated by the loss of all but one AC power source to 4160V essential buses 64B/64C (Division 1) and 65E/65F (Division II) for greater then or equal to 15 minutes such that a loss of any additional source will result in a complete loss of AC power to essential busses.

The safety-related essential electrical AC system consists of two physically and electrically independent and redundant power trains, Division I and Division II, supplying electrical power to safety-related equipment. Either Division I or Division II has the capability and the capacity to supply the essential AC power loads in the respective division. Each division is equipped with a split bus, radially fed network including four 4160-volt buses, four 480-volt buses and associated transformers. For the purpose of emergency classification, the loss of ability to power either split bus in a division (64B or C OR 65E or F) is considered an inability to power the respective division. Similarly, a loss of either EDG in a division is considered a loss of ability to power that divisions essential buses. Division I is normally supplied with offsite power from the 120-kV switchyard, whereas Division II is normally supplied from the 345-kV switchyard, thus providing two physically and electrically independent offsite sources (Ref. 1).

CTG 11-1 (alternatively CTGs 11-2, 11-3 or 11-4) provides a 120 kV AC line to a 13.8 kV/4160 V transformer to the essential buses. This Alternate AC (AAC) Power Supply has the capacity to supply only one fully loaded essential division (Division 1 unless cross-tied) within 1 hour (Ref. 2). Credit can be taken for CTG 11-1 (alternatively CTGs 11-2, 11-3 or 11-4) as an onsite AC power supply only if it is already aligned to and capable of powering one of the essential 4160 V divisions within the 15 minute time criteria (Ref. 2).

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

This cold condition EAL is equivalent to the hot condition loss of all offsite AC power EAL SA1.1.

Generic

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 essential bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of one division of essential power sources (e.g., onsite diesel generators).
- A loss of essential power sources (e.g., onsite diesel generators) with a single division of essential 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.

Fermi Basis Reference(s):

1. Design Basis Document RXX-00 ESS Electrical AC Systems
2. UFSAR Chapter 8 Electrical Power
3. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Essential 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 (Table C-2) to 4160V essential Division I (64B/64C) and Division II (65E/65F) for ≥ 15 min. (Note 1, 10)

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

Note 10: Credit may be taken for one of the four CTGs as an onsite AC power source only if the CTG is already aligned and capable of powering an essential bus within 15 min.

Table C-2 AC Power Sources
Offsite: • System Service Transformer 64 (Div I) • System Service Transformer 65 (Div II) **Onsite:** • EDGs 11/12 (Div I) • EDGs 13/14 (Div II) • CTG 11-1, 11-2, 11-3 or 11-4

Mode Applicability:

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

Definition(s):

None

Basis:

Plant-Specific

Table C-2 lists AC sources capable of powering essential AC divisions. For emergency classification purposes, “capability” means that an AC power source is available to the essential divisional buses, whether or not the buses are currently powered from it.

This EAL is indicated by the loss of all offsite and onsite AC power capability to 4160V essential buses 64B/64C (Division I) and 65E/65F (Division II) for greater then or equal to 15 minutes.

The safety-related essential electrical AC system consists of two physically and electrically independent and redundant power trains, Division I and Division II, supplying electrical power to safety-related equipment. Either Division I or Division II has the capability and the capacity to supply the essential AC power loads in the respective division. Each division is equipped with a split bus, radially fed network including four 4160-volt buses, four 480-volt buses and associated transformers. For the purpose of emergency classification, the loss of ability to power either split bus in a division (64B or C <u>OR</u> 65E or F) is considered an inability to power the respective division. Similarly, a loss of either EDG in a division is considered a loss of ability to power that divisions essential buses. Division I is normally supplied with offsite power from the 120-kV switchyard, whereas Division II is normally supplied from the 345-kV switchyard, thus providing two physically and electrically independent offsite sources (Ref. 1).

CTG 11-1 (alternatively CTGs 11-2, 11-3 or 11-4) provides a 120 kV AC line to a 13.8 kV/4160 V transformer to the essential buses. This Alternate AC (AAC) Power Supply has the capacity to supply only one fully loaded essential division (Division 1 unless cross-tied) within 1 hour (Ref. 2). Credit can be taken for CTG 11-1 (alternatively CTGs 11-2, 11-3 or 11-4) as an onsite AC power supply only if it is already aligned to and capable of powering one of the essential 4160 V divisions within the 15 minute time criteria (Ref. 2).

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

<u>Generic</u>

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as a Site Area Emergency because of the increased time available to restore an essential 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.

Fermi Basis Reference(s):

1. Design Basis Document RXX-00 ESS Electrical AC Systems
2. UFSAR Chapter 8 Electrical Power
3. NEI 99-01 CA2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature.

EAL:

CU3.1	**Unusual Event**
UNPLANNED increase in RCS temperature to > 200°F	

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.

Basis:

Plant-Specific

Several instruments are capable of providing indication of RCS temperature (Ref. 2) with respect to the Technical Specification cold shutdown temperature limit (200°F, Ref. 1):

Primary:

- Recirc Loop A(B) Suction Temperature - B31-R650 (H11-P603)
- Reactor Vessel Shell Temperature - B21-R007 (H11-P603)
- Reactor Vessel Bottomhead Drain - G33-N601 (H11-P602) (drain line must have flow)

Alternate:

- RHR A/B HX Inlet - E11-R601A/B
- Reactor Vessel Shell Temperature - B21-R007 (H11-P603)

Generic

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 should also refer to IC CA3.

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

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 CA3 based on exceeding plant configuration-specific time criteria.

Fermi Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. GOP 22.000.05 Pressure/Temperature Monitoring During Heatup and Cooldown
3. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature.

EAL:

CU3.2	**Unusual Event**
Loss of **all** RCS temperature and RPV level indication for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director 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

Basis:

Plant-Specific

Several instruments are capable of providing indication of RCS temperature (Ref. 2) with respect to the Technical Specification cold shutdown temperature limit (200°F, Ref. 1):

- Primary:
  - Recirc Loop A(B) Suction Temperature - B31-R650 (H11-P603)
  - Reactor Vessel Shell Temperature - B21-R007 (H11-P603)
  - Reactor Vessel Bottomhead Drain - G33-N601 (H11-P602) (drain line must have flow)
- Alternate:
  - RHR A/B HX Inlet - E11-R601A/B
  - Reactor Vessel Shell Temperature - B21-R007 (H11-P603)

RPV level can be monitored by one or more of the following (Ref. 3):

- Flood-up Level indicator B21-R605 (+160 in. to +560 in.)
- Wide Range Level indicators B21-R604A/B (+10 in. to +220 in.)

- Narrow Range Level indicators C32-R606A/B/C/D (+160 in. to +220 in.)
- Floodup Level Transmitter (when in service)

  During refueling operations with the fuel pool gates removed, the RPV floodup level instrumentation (B21-N027) and the Rx Vessel Core Plate dp transmitter (B21-N032) are capable of displaying the common level of the reactor cavity and the spent fuel pool. (Ref. 4)

Generic

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 should also refer to IC CA3.

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 CA3 based on exceeding plant configuration-specific time criteria.

Fermi Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. GOP 22.000.05 Pressure/Temperature Monitoring During Heatup and Cooldown
3. UFSAR Section 7 Instrumentation and Controls Table 7.5-1 Control Room Level Indication
4. SOP 23.800.06 Rev. 10 Reactor Vessel Water Level Monitoring During Refueling Operations
5. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

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

EAL:

CA3.1 Alert

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

Note 1: The Emergency Director should declare the event promptly upon determining that time limit 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 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.

For Fermi 2, this condition is met if either Primary Containment or Secondary Containment are functional (i.e. intact).

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.

Basis:

Plant-Specific

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 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 or nozzle dams).

Several instruments are capable of providing indication of RCS temperature (Ref. 2) with respect to the Technical Specification cold shutdown temperature limit (200°F, Ref. 1):

Primary:

- Recirc Loop A(B) Suction Temperature - B31-R650 (H11-P603)
- Reactor Vessel Shell Temperature - B21-R007 (H11-P603)
- Reactor Vessel Bottomhead Drain - G33-N601 (H11-P602) (drain line must have flow)

Alternate:

- RHR A/B HX Inlet - E11-R601A/B
- Reactor Vessel Shell Temperature - B21-R007 (H11-P603)

Generic

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.

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

Fermi Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. GOP 22.000.05 Pressure/Temperature Monitoring During Heatup and Cooldown
3. NEI 99-01 CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

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

EAL:

CA3.2	**Alert**
UNPLANNED RPV pressure increase > 10 psig	

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.

Basis:

Plant-Specific

A 10 psig RPV pressure increase can be monitored on various indicators such as C32-R609 (H11-P603) (Ref. 1).

Generic

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.

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

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

Fermi Basis Reference(s):

1. GOP 22.000.05 Pressure/Temperature Monitoring During Heatup and Cooldown
2. NEI 99-01 CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – Loss of Vital DC Power

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

EAL:

CU4.1	**Unusual Event**
Degraded voltage (< 105 VDC) on **required** 130 VDC system vital buses for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director 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

Basis:

Plant-Specific

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.

The fifteen minute interval is intended to exclude transient or momentary power losses.

At Fermi 2, the vital 260/130 VDC System ensures power is available for the reactor to be shutdown safely and maintained in a safe condition. The vital DC system is divided into two independent divisions - Division I and Division II - with separate DC power supplies. These power supplies consist of two separate 260/130V batteries and chargers serving systems such as RCIC, RHR, EDGs, and HPCI. The system provides sufficient capacity, via each of the Class 1E DC batteries, to power all required loads for 4 hours following a loss of AC power supply (Ref. 1).

Based on Technical Specifications Bases Section B.3.8.4, the 130 VDC battery minimum design voltage limit is 105 VDC. (Ref. 2).

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

Generic

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 CA3, or an IC in Recognition Category R.

Fermi Basis Reference(s):

1. Design Bases Document R32-00 DC Electrical System
2. Technical Specifications Bases Section B.3.8.4 DC Sources - Operating
3. NEI 99-01 CU4

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 offsite communication methods

OR

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

Table C-4 Communication Methods			
System	**Onsite**	**Offsite**	**NRC**
Administrative Telephones	X	X	X
RERP Emergency Telephones	X	X	X
Satellite Phones		X	X
Federal Telephone System (ENS)		X	X
Automatic Ring Lines		X	
MI State Radios (800 MHz)		X	
Plant Radio System	X		
Hi-Com (PA System)	X		

Mode Applicability:

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

Definition(s):

None

Basis:

Plant-Specific

The Table C-4 list for onsite communications loss encompasses the loss of all means of routine communications (e.g., administrative and internal telephones, plant page [Hi-Com] and plant radios) (Ref. 1, 2).

The Table C-4 list for offsite communications loss encompasses the loss of all means of communications with offsite authorities. This includes the RERP telephone dedicated ring lines, backup phone systems administrative telephone lines, satellite, and FTS (ENS) which can be utilized as a regular telephone (Ref. 1, 2).

The Table C-4 list for NRC communications loss encompasses the loss of all means of communications with the NRC. This includes the FTS (ENS), backup phone systems (administrative telephone lines, RERP phones and satellite) (Ref. 1, 2).

The communications methods used at Fermi 2 are described in the RERP Plan (Ref. 1).

The radio network at Fermi 2 involves several radio systems to effect communications within the plant with damage control teams, rescue teams, fire brigade, radiological monitoring teams, and security personnel as well as provide backup communications to essential Offsite Emergency Response Organizations (OROs) in the event of telephone equipment malfunction.

There are two radio consoles normally used in the Control Room. One is installed in panel H11-P700 to establish communications using plant radio zone 1 (control room group) to hand-held portable radios (OPS channel 1 or 2) via the plant radio repeater system.

An additional radio console is located in panel H11-P703 to allow for backup communications to hand-held portable radios on various other user groups via plant radio zone 1 repeater system or backup repeaters (zone 2). Maintenance channels 1, 2, or 3 can also be selected at this station. This console also provides a backup radio communication selection into security zone 3 that provides another two repeaters for radio operation.

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

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

Generic

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 State, Monroe and Wayne County EOCs.

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

Fermi Basis Reference(s):

1. Fermi Emergency Plan Section F Emergency Communications
2. EP-580 Equipment Important to Emergency Response
3. 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 required for the current operating mode.

EAL:

CA6.1 Alert

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

AND

EITHER of the following:

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

Basis:

<u>Plant-Specific</u>

- The significance of seismic events are discussed under EAL HU2.1 (Ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (Ref. 2, 3).
- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 90 mph (sustained). (Ref. 4).

- Areas containing functions and systems required for safe shutdown of the plant are identified by Fire Zone in the fire response procedure (Ref. 5).
- 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.

Generic

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.

Fermi Basis Reference(s):

1. AOP 20.000.01 Acts of Nature
2. AOP 20.000.03 Turbine Building Flooding
3. PLG-0849 Fermi 2 Internal Flooding Analysis
4. UFSAR Section 3.3.3.1 Design Wind Speed
5. AOP 20.000.22 Plant Fires
6. NEI 99-01 CA6

Category H – Hazards and Other Conditions Affecting Plant Safety

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

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

The events of this category pertain to the following subcategories:

1. 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 Technological 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

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

7. ED Judgment

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

Category: H – Hazards and Other Conditions Affecting Plant Safety

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 Security Shift Supervisor

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 Fermi 2 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 Fermi 2. 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).

Basis:

Plant-Specific

If the Security Shift Supervisor determines that a threat notification is credible, the Security Shift Supervisor will notify the Shift Manager that a “Credible Threat” condition exists for

Fermi 2. The three main criteria for determining credibility are: technical feasibility, operational feasibility, and resolve. For Fermi 2, a validated notification delivered by the FBI, NRC or similar agency is treated as credible (Ref. 1, 2).

Generic

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

The second threshold addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the Fermi Safeguards Contingency 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 Fermi Safeguards Contingency 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 Fermi Safeguards Contingency Plan (Ref. 1).

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

Fermi Basis Reference(s):

1. Fermi Safeguards Contingency Plan
2. NEI 99-01 HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

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 Supervisor

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 Fermi 2 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 Fermi 2. 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 - The company property immediately surrounding the PROTECTED AREA security fence. Access is normally limited to people on official business.

Basis:

Plant-Specific

The Owner Controlled Area is depicted in the Fermi 2 Radiological Emergency Response Preparedness Plan Figure J-1 Owner-Controlled Area (Ref. 1).

Generic

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 Supervisor 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.

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 Fermi Safeguards Contingency Plan (Ref. 2).

Escalation of the emergency classification level would be via IC HS1.

Fermi Basis Reference(s):

1. Fermi 2 Radiological Emergency Response Preparedness Plan Figure J-1 Owner-Controlled Area
2. Fermi Safeguards Contingency Plan
3. NEI 99-01 HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

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 Supervisor

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward Fermi 2 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 Fermi 2. 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 - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in UFSAR Figure 1.2-5 Site Plot Plan (Ref. 1).

Basis:

Plant-Specific

None

Generic

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 Supervisor 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 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 Fermi Safeguards Contingency Plan (Ref. 2).

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

Fermi Basis Reference(s):

1. UFSAR Figure 1.2-5 Site Plot Plan
2. Fermi Safeguards Contingency Plan
3. NEI 99-01 HS1

Category: H – Hazards and Other Conditions Affecting Plant Safety

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 Supervisor

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 Fermi 2 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 Fermi 2. 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 - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in UFSAR Figure 1.2-5 Site Plot Plan (Ref. 1).

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.

Basis:

Plant-Specific

None

Generic

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 Supervisor 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 Fermi Safeguards Contingency Plan (Ref. 2).

Fermi Basis Reference(s):

1. UFSAR Figure 1.2-5 Site Plot Plan
2. Fermi Safeguards Contingency Plan
3. NEI 99-01 HG1

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 peak accelerations > 0.05g vertical or > 0.08g horizontal on D30-R800 Active Seismic Playback Printer	

Mode Applicability:

All

Definition(s):

None

Basis:

Plant-Specific

The instrumentation used to indicate a seismic event includes the Triaxial Seismic Switch and the Triaxial Response Spectrum Recorder. Annunciator, ARP 6D69 (SEISMIC SYSTEM EVENT/TROUBLE), is sounded in the Control Room whenever the Triaxial Seismic Switch senses ground acceleration in excess of 0.01g (Ref. 1, 2, 3, 4). The Fermi 2 seismic instrumentation supports readily assessable (within 15 minutes) OBE indications (> 0.08g horizontal, > 0.05g vertical acceleration).

An offsite agency (USGS, National Earthquake Information Center) can confirm that an earthquake has occurred in the area of the plant. However, such confirmation should not preclude timely emergency declaration. Provide the analyst with the following Fermi 2 coordinates: **41º 57' 48" north latitude, 83º 15' 31" west longitude** (Ref. 5).

Generic

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 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.

Fermi Basis Reference(s):

1. UFSAR Section 3.7.4 Seismic Instrumentation Program
2. AOP 20.000.01 Acts of Nature
3. ARP 6D69 Seismic System Event/Trouble
4. SOP 23.612 Seismic Monitoring
5. UFSAR Section 1.2.2.1 Location and Size of Site
6. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

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

Mode Applicability:

All

Definition(s):

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in UFSAR Figure 1.2-5 Site Plot Plan (Ref. 1).

Basis:

Plant-Specific

None

Generic

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.

Fermi Basis Reference(s):

1. UFSAR Figure 1.2-5 Site Plot Plan
2. AOP 20.000.01 Acts of Nature
3. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological 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 required 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.

Basis:

Plant-Specific

The term “required’ as used in this EAL is defined as the number of operable systems required by Technical Specifications for the present operating mode. Therefore, isolation of components that do not affect the required number of systems required to meet Technical Specifications for the current mode would not require classification.

Refer to Fermi 2 "Internal Flooding Analysis" to identify susceptible internal Flooding Areas (Ref. 2).

Generic

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. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

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

Fermi Basis Reference(s):

1. AOP 20.000.03 Turbine Building Flooding
2. PLG-0849 Fermi 2 Internal Flooding Analysis
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological 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):

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in UFSAR Figure 1.2-5 Site Plot Plan (Ref. 1).

Basis:

Plant-Specific

As used here, the term "offsite" is meant to be areas external to the Fermi 2 PROTECTED AREA.

AOP 20.000.30, "Offsite Release of Toxic/Flammable Gas", provides additional information on hazardous substances and spills. Potential sources of toxic gases are chlorine and anahydrous ammonia (Ref. 2).

Generic

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.

Fermi Basis Reference(s):

1. UFSAR Figure 1.2-5 Site Plot Plan
2. AOP 20.000.30 Offsite Release of Toxic/Flammable Gas
3. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological 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

Basis:

Plant-Specific

None

Generic

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.

Fermi 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 Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Reactor Building • Auxiliary Building • Reactor Building Steam Tunnel • RHR Complex • 4160V Ductbanks between RHR Complex and Auxiliary Building

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.

Basis:

Plant-Specific

Table H-1 Fire Areas are based on UFSAR Section 3.2 Classification of Structures, Components and Systems. Category I structures are those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (Ref. 1).

Generic

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 emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

Basis-Related Requirements from Appendix R

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."
>
> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.

> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in this EAL, the 15-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.

Fermi Basis Reference(s):

1. UFSAR Section 3.2 Classification of Structures, Components and Systems
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 within 30 min. of alarm receipt (Note 1)

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

Table H-1 Fire Areas
• Reactor Building • Auxiliary Building • Reactor Building Steam Tunnel • RHR Complex • 4160V Ductbanks between RHR Complex and Auxiliary Building

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.

Basis:

Plant-Specific

Table H-1 Fire Areas are based on UFSAR Section 3.2 Classification of Structures, Components and Systems. Category I structures are those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (Ref. 1).

Generic

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.

Basis-Related Requirements from Appendix R

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize,

consistent with other safety requirements, the probability and effect of fires and explosions."

When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.

Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in this EAL, 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.

Fermi Basis Reference(s):

1. UFSAR Section 3.2 Classification of Structures, Components and Systems
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 Emergency Director 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 - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in UFSAR Figure 1.2-5 Site Plot Plan (Ref. 1).

Basis:

<u>Plant-Specific</u>

None

<u>Generic</u>

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.

Fermi Basis Reference(s):

1. UFSAR Figure 1.2-5 Site Plot Plan
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:

HU4.4	**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 - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in UFSAR Figure 1.2-5 Site Plot Plan (Ref. 1).

Basis:

Plant-Specific

None

Generic

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.

Fermi Basis Reference(s):

1. UFSAR Figure 1.2-5 Site Plot Plan
2. 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 Shutdown Rooms/Areas

Room/Area	Mode Applicability
• Control Room	All
• Relay Room	All
• AB3-DC MCC Area	Mode 3
• RB1-F17	Mode 3
• RB1-F11	Mode 3

Mode Applicability:

All

Definition(s):

None

Basis:

Plant-Specific

The rooms/areas and associated mode applicability specified in Table H-2 are those that contain equipment which require a manual/local action as specified in operating procedures used for normal operation, cooldown and shutdown. This table excludes rooms/areas that may have procedurally directed actions that are of an administrative nature (normal rounds or routine inspections) or that are not crucial to the conduct of safe operation, cooldown and shutdown.

Specifically:

- Control Room & Relay Room in all modes
- AB3-DC MCC Area – Access is required when in Mode 3 to install power fuses and close the MCC for E1150-F008 which must be performed to align shutdown cooling suction path.
- RB1-F17 – Access is required when in Mode 3 to place the permissive switch in OPERATE for E11F610A if Div 1 RHR is being placed in shutdown cooling. This step must be performed to warmup the shutdown cooling piping.
- RB1-F11 – Access is required when in Mode 3 to place the permissive switch in OPERATE for E11F610B if Div 2 RHR is being placed in shutdown cooling. This step must be performed to warmup the shutdown cooling piping.

(Ref. 1, 2, 3).

<u>Generic</u>

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

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

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

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

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 3.
- 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 normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

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

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

Fermi Basis Reference(s):

1. Operating Procedures (normal plant operations, cooldown or shutdown), manual / local actions:
   a. 22.000.03 - Power Operation 25% to 100% to 25%
   b. 22.000.04 - Plant Shutdown From 25% Power
   c. 22.000.05 - Pressure/Temp Monitoring During Heatup and Cooldown
   d. 23.202 - High Pressure Coolant Injection System
   e. 23.205 - Residual Heat Removal System
   f. 23.206 - Reactor Core Isolation Cooling System
   g. 23.427 - Primary Containment Isolation System
   h. 23.610 - Reactor Protection System (RPS)
   i. MGA03 - Procedure Use and Adherence
2. GOP 22.000.03 Power Operation 25% to 100% to 25%
3. GOP 22.000.04 Plant Shutdown from 25% Power
4. NEI 99-01 HA5

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 Dedicated or Remote Shutdown Panels	

Mode Applicability:

All

Definition(s):

None

Basis:

Plant-Specific

For the purpose of this EAL the 15 minute classification clock starts when the last licensed operator leaves the Control Room.

Per AOP 20.000.18, "Control of the Plant from the Dedicated Shutdown Panel", (Ref. 1) and AOP 20.000.19, "Shutdown from Outside the Control Room", (Ref. 2) plant control is established at the Dedicated or Remote Shutdown Panels when:

- Initial Control Room actions are complete
- All available Operators have reported to the Dedication or Remote Shutdown Panels
- RPV level and pressure are being controlled from the Dedicated or Remote Shutdown Panels

Generic

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.

Fermi Basis Reference(s):

1. AOP 20.000.18 Control of the Plant from the Dedicated Shutdown Panel
2. AOP 20.000.19 Shutdown from Outside the Control Room
3. UFSAR Section 7.5.1.5.5 Procedure for Reactor Shutdown from Outside the Main Control Room
4. NEI 99-01 HA6

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 Dedicated or Remote Shutdown Panels

AND

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

- Reactivity (Mode 1 and 2 **only**)
- RPV water level
- RCS heat removal

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 – Cold Shutdown, 5 - Refueling

Definition(s):

None

Basis:

Plant-Specific

For the purpose of this EAL the 15 minute clock starts when the last licensed operator leaves the Control Room.

Per AOP 20.000.18, "Control of the Plant from the Dedicated Shutdown Panel", (Ref. 1) and AOP 20.000.19, "Shutdown from Outside the Control Room" (Ref. 2) plant control is established at the Dedicated or Remote Shutdown Panels when:

- Initial Control Room actions are complete
- All available Operators have reported to the Dedicated or Remote Shutdown Panels

- RPV level and pressure are being controlled from the Dedicated or Remote Shutdown Panels

<u>Generic</u>

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 judgment. The Emergency Director is expected to make a reasonable, informed judgment within 15 minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

Escalation of the emergency classification level would be via IC FG1 or CG1

Fermi Basis Reference(s):

1. AOP 20.000.18 Control of the Plant from the Dedicated Shutdown Panel
2. AOP 20.000.19 Shutdown from Outside the Control Room
3. UFSAR Section 7.5.1.5.5 Procedure for Reactor Shutdown from Outside the Main Control Room
4. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Judgment

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

EAL:

HU7.1 Unusual Event

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

Basis:

<u>Plant-Specific</u>

None

Generic

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

Fermi Basis Reference(s):

1. NEI 99-01 HU7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Judgment

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

EAL:

HA7.1	**Alert**
Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward Fermi 2 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 Fermi 2. 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).

Basis:

Plant-Specific

None

Generic

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

Fermi Basis Reference(s):

1. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Judgment

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

EAL:

HS7.1 Site Area Emergency

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the SITE BOUNDARY.

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward Fermi 2 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 Fermi 2. 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)

SITE BOUNDARY - That line beyond which the land is neither owned, nor leased, nor otherwise controlled by DTE Electric.

Basis:

Plant-Specific

None

<u>Generic</u>

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

Fermi Reference(s):

1. NEI 99-01 HS7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Judgment

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

EAL:

HG7.1	**General Emergency**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve 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 Fermi 2 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 Fermi 2. 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.

Basis:

<u>Plant-Specific</u>

None

Generic

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

Fermi Basis Reference(s):

1. NEI 99-01 HG7

Category S – System Malfunction

EAL Group: Hot Conditions (RCS temperature > 200ºF); 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 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 4160 VAC essential buses 64B/64C and 65E/65F.

2. 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 130 VDC ESF 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 Category F, Fission Product Barrier Degradation. 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

Various natural and technological events that result in degraded plant safety system performance or significant visible damage warrant emergency classification under this subcategory.

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 (Table S-1) to 4160V essential Division I (64B/64C) and Division II (65E/65F) for ≥ 15 min. (Note 1, 10)

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

Note 10: Credit may be taken for one of the four CTGs as an onsite AC power source only if the CTG is already aligned and capable of powering an essential bus within 15 min.

Table S-1 AC Power Sources
Offsite: • System Service Transformer 64 (Div I) • System Service Transformer 65 (Div II) **Onsite:** • EDGs 11/12 (Div I) • EDGs 13/14 (Div II) • CTG 11-1, 11-2, 11-3 or 11-4

Mode Applicability:

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

Definition(s):

None

Basis:

Plant-Specific

Table S-1 lists AC sources capable of powering essential buses 64B/64C (Division 1) and 65E/65F (Division 2). For emergency classification purposes, “capability” means that an AC power source is available to the essential divisional buses, whether or not the buses are currently powered from it.

This EAL is indicated by the loss of capability of all offsite AC power sources to power 4160V essential Division I (64B/64C) and Division II (65E/65F) for greater then or equal to 15 minutes.

The safety-related essential electrical AC system consists of two physically and electrically independent and redundant power trains, Division I and Division II, supplying electrical power to safety-related equipment. Either Division I or Division II has the capability and the capacity to supply the essential AC power loads in the respective division. Each division is equipped with a split bus, radially fed network including four 4160-volt buses, four 480-volt buses and associated transformers. For the purpose of emergency classification, the loss of ability to power either split bus in a division (64B or C OR 65E or F) is considered an inability to power the respective division. Similarly, a loss of either EDG in a division is considered a loss of ability to power that divisions essential buses. Division I is normally supplied with offsite power from the 120-kV switchyard, whereas Division II is normally supplied from the 345-kV switchyard, thus providing two physically and electrically independent offsite sources (Ref. 1).

CTG 11-1 (alternatively CTGs 11-2, 11-3 or 11-4) provides a 120 kV AC line to a 13.8 kV/4160 V transformer to the essential buses. This Alternate AC (AAC) Power Supply has the capacity to supply only one fully loaded essential division (Division 1 unless cross-tied) within 1 hour (Ref. 2). Credit can be taken for CTG 11-1 (alternatively CTGs 11-2, 11-3 or 11-4) as an onsite AC power supply only if it is already aligned to and capable of powering one of the essential 4160 V divisions within the 15 minute time criteria (Ref. 2).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses.

Generic

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 essential 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 essential 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.

Fermi Basis Reference(s):

1. Design Basis Document RXX-00 ESS Electrical AC Systems
2. UFSAR Chapter 8 Electrical Power
3. NEI 99-01 SU1

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 4160V essential Division I (64B/64C) and Division II (65E/65F) reduced to a single power source (Table S-1) for ≥ 15 min. (Note 1, 10)

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

Note 10: Credit may be taken for one of the four CTGs as an onsite AC power source only if the CTG is already aligned and capable of powering an essential bus within 15 min.

Table S-1 AC Power Sources
Offsite:
• System Service Transformer 64 (Div I)
• System Service Transformer 65 (Div II)
Onsite:
• EDGs 11/12 (Div I)
• EDGs 13/14 (Div II)
• CTG 11-1, 11-2, 11-3 or 11-4

Mode Applicability:

1 - Power Operation, 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:

Plant-Specific

Table S-1 lists AC sources capable of powering essential buses. For emergency classification purposes, “capability” means that an AC power source is available to the essential divisional buses, whether or not the buses are currently powered from it.

This EAL is indicated by the loss of all but one AC power source to 4160V essential buses 64B/64C (Division I) and 65E/65F (Division II) for greater then or equal to 15 minutes such that a loss of any additional source will result in a complete loss of AC power to essential busses.

The safety-related essential electrical AC system consists of two physically and electrically independent and redundant power trains, Division I and Division II, supplying electrical power to safety-related equipment. Either Division I or Division II has the capability and the capacity to supply the essential AC power loads in the respective division. Each division is equipped with a split bus, radially fed network including four 4160-volt buses, four 480-volt buses and associated transformers. For the purpose of emergency classification, the loss of ability to power either split bus in a division (64B or C OR 65E or F) is considered an inability to power the respective division. Similarly, a loss of either EDG in a division is considered a loss of ability to power that divisions essential buses. Division I is normally supplied with offsite power from the 120-kV switchyard, whereas Division II is normally supplied from the 345-kV switchyard, thus providing two physically and electrically independent offsite sources (Ref. 1).

CTG 11-1 (alternatively CTGs 11-2, 11-3 or 11-4) provides a 120 kV AC line to a 13.8 kV/4160 V transformer to the essential buses. This Alternate AC (AAC) Power Supply has the capacity to supply only one fully loaded essential division (Division 1 unless cross-tied) within 1 hour (Ref. 2). Credit can be taken for CTG 11-1 (alternatively CTGs 11-2, 11-3 or

11-4) as an onsite AC power supply only if it is already aligned to and capable of powering one of the essential 4160 V divisions within the 15 minute time criteria (Ref. 2).

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

Generic

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 essential bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of one essential power source (e.g., an onsite diesel generators).
- A loss of essential power sources (e.g., onsite diesel generators) with a single division of essential 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.

Fermi Basis Reference(s):

1. Design Basis Document RXX-00 ESS Electrical AC Systems
2. UFSAR Chapter 8 Electrical Power
3. NEI 99-01 SA1

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of all offsite 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 (Table S-1) to 4160V essential Division I (64B/64C) and Division II (65E/65F) for ≥ 15 min. (Note 1, 10)

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

Note 10: Credit may be taken for one of the four CTGs as an onsite AC power source only if the CTG is already aligned and capable of powering an essential bus within 15 min.

Table S-1 AC Power Sources
Offsite: • System Service Transformer 64 (Div I) • System Service Transformer 65 (Div II) **Onsite:** • EDGs 11/12 (Div I) • EDGs 13/14 (Div II) • CTG 11-1, 11-2, 11-3 or 11-4

Mode Applicability:

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

Definition(s):

None

Basis:

Plant-Specific

Table S-1 lists AC sources capable of powering essential Division I and Division II AC buses. For emergency classification purposes, “capability” means that an AC power source is available to the essential divisional buses, whether or not the buses are currently powered from it.

This EAL is indicated by the loss of all offsite and onsite AC power capability to 4160V essential Division I buses 64B/64C and Division II 65E/65F for greater then or equal to 15 minutes.

The safety-related essential electrical AC system consists of two physically and electrically independent and redundant power trains, Division I and Division II, supplying electrical power to safety-related equipment. Either Division I or Division II has the capability and the capacity to supply the essential AC power loads in the respective division. Each division is equipped with a split bus, radially fed network including four 4160-volt buses, four 480-volt buses and associated transformers. For the purpose of emergency classification, the loss of ability to power either split bus in a division (64B or C OR 65E or F) is considered an inability to power the respective division. Similarly, a loss of either EDG in a division is considered a loss of ability to power that divisions essential buses. Division I is normally supplied with offsite power from the 120-kV switchyard, whereas Division II is normally supplied from the 345-kV switchyard, thus providing two physically and electrically independent offsite sources (Ref. 1).

CTG 11-1 (alternatively CTGs 11-2, 11-3 or 11-4) provides a 120 kV AC line to a 13.8 kV/4160 V transformer to the essential buses. This Alternate AC (AAC) Power Supply has the capacity to supply only one fully loaded essential division (Division 1 unless cross-tied) within 1 hour (Ref. 2). Credit can be taken for CTG 11-1 (alternatively CTGs 11-2, 11-3 or 11-4) as an onsite AC power supply only if it is already aligned to and capable of powering one of the essential 4160 V divisions within the 15 minute time criteria (Ref. 2).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses.

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

<u>Generic</u>

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. 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.

Fermi Basis Reference(s):

1. Design Basis Document RXX-00 ESS Electrical AC Systems
2. UFSAR Chapter 8 Electrical Power
3. NEI 99-01 SS1

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 OR loss of all essential AC and vital DC power sources for 15 minutes or longer.

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power capability (Table S-1) to 4160V essential Division I (64B/64C) and Division II (65E/65F) (Note 10)

AND EITHER of the following:

- Restoration of at least one essential division within 4 hours is **not** likely (Note 1)
- RPV water level CANNOT be restored and maintained > -25 in.

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

Note 10: Credit may be taken for one of the four CTGs as an onsite AC power source only if the CTG is already aligned and capable of powering an essential bus within 15 min.

Table S-1 AC Power Sources
Offsite: • System Service Transformer 64 (Div I) • System Service Transformer 65 (Div II) **Onsite:** • EDGs 11/12 (Div I) • EDGs 13/14 (Div II) • CTG 11-1, 11-2, 11-3 or 11-4

Mode Applicability:

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

Definition(s):

None

Basis:

Plant-Specific

Table S-1 lists AC sources capable of powering essential AC buses. For emergency classification purposes, “capability” means that an AC power source is available to the essential divisional buses, whether or not the buses are currently powered from it (Ref. 1, 2).

This EAL is indicated by the extended loss of offsite and onsite AC power capability to 4160V essential Division I buses 64B/64C and Division II buses 65E/65F either for greater then the Fermi 2 Station Blackout (SBO) coping analysis time (4 hrs.) (Ref. 3) or that has resulted in indications of an actual loss of adequate core cooling.

Indication of continuing core cooling degradation is manifested by the inability to restore and maintain RPV water level above the Minimum Steam Cooling RPV Water Level (Ref. 4).

The safety-related essential electrical AC system consists of two physically and electrically independent and redundant power trains, Division I and Division II, supplying electrical power to safety-related equipment. Either Division I or Division II has the capability and the capacity to supply the essential AC power loads in the respective division. Each division is equipped with a split bus, radially fed network including four 4160-volt buses, four 480-volt buses and associated transformers. For the purpose of emergency classification, the loss of ability to power either split bus in a division (64B or C <u>OR</u> 65E or F) is considered an inability to power the respective division. Similarly, a loss of either EDG in a division is considered a loss of ability to power that divisions essential buses. Division I is normally supplied with offsite power from the 120-kV switchyard, whereas Division II is normally supplied from the 345-kV switchyard, thus providing two physically and electrically independent offsite sources (Ref. 1).

CTG 11-1 (alternatively CTGs 11-2, 11-3 or 11-4) provides a 120 kV AC line to a 13.8 kV/4160 V transformer to the essential buses. This Alternate AC (AAC) Power Supply has the capacity to supply only one fully loaded essential division (Division 1 unless cross-tied) within 1 hour (Ref. 2).

<u>Generic</u>

This IC addresses a prolonged loss of all power sources to AC essential 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 4160V essential 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 4160V essential 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.

Fermi Basis Reference(s):

1. Design Basis Document RXX-00 ESS Electrical AC Systems
2. UFSAR Chapter 8 Electrical Power
3. UFSAR Section 8.4.2.1 SBO Coping Duration
4. EOP Support Documentation Section 1 Plant Specific Technical Guideline (PSTG)
5. NEI 99-01 SG1

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 OR 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 (Table S-1) to 4160V essential Division I (64B/64C) and Division II (65E/65F) for ≥ 15 min. (Note 1, 10)

AND

Degraded voltage (< 105 VDC) on **both** 130 VDC system vital buses for ≥ 15 min. (Note 1)

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

Note 10: Credit may be taken for one of the four CTGs as an onsite AC power source only if the CTG is already aligned and capable of powering an essential bus within 15 min.

Table S-1 AC Power Sources
Offsite: • System Service Transformer 64 (Div I) • System Service Transformer 65 (Div II) **Onsite:** • EDGs 11/12 (Div I) • EDGs 13/14 (Div II) • CTG 11-1, 11-2, 11-3 or 11-4

Mode Applicability:

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

Definition(s):

None

Basis:

Plant-Specific

Table S-1 lists AC sources capable of powering essential AC divisions. For emergency classification purposes, “capability” means that an AC power source is available to the

essential divisional buses, whether or not the buses are currently powered from it (Ref. 1, 2).

This EAL is indicated by the loss of all offsite and onsite essential AC power capability to 4160V essential Division I (64B/64C) and Division II (65E/65F) for greater than 15 minutes in combination with degraded vital DC power voltage. This EAL addresses operating experience from the March 2011 accident at Fukushima Daiichi.

The safety-related essential electrical AC system consists of two physically and electrically independent and redundant power trains, Division I and Division II, supplying electrical power to safety-related equipment. Either Division I or Division II has the capability and the capacity to supply the essential AC power loads in the respective division. Each division is equipped with a split bus, radially fed network including four 4160-volt buses, four 480-volt buses and associated transformers. For the purpose of emergency classification, the loss of ability to power either split bus in a division (64B or C <u>OR</u> 65E or F) is considered an inability to power the respective division. Similarly, a loss of either EDG in a division is considered a loss of ability to power that divisions essential buses. Division I is normally supplied with offsite power from the 120-kV switchyard, whereas Division II is normally supplied from the 345-kV switchyard, thus providing two physically and electrically independent offsite sources (Ref. 1).

CTG 11-1 provides a 120 kV AC line to a 13.8 kV/4160 V transformer to the essential buses. This Alternate AC (AAC) Power Supply has the capacity to supply only one fully loaded essential division within 1 hour (Ref. 2). Credit can be taken for CTG 11-1 as an onsite AC power supply only if it is already aligned to one of the essential 4160 V divisions.

At Fermi 2, the vital 260/130 VDC System ensures power is available for the reactor to be shutdown safely and maintained in a safe condition. The vital DC system is divided into two independent divisions - Division I and Division II - with separate DC power supplies. These power supplies consist of two separate 260/130V batteries and chargers serving systems such as RCIC, RHR, EDGs, and HPCI. The system provides sufficient capacity, via each of the Class 1E DC batteries, to power all required loads for 4 hours following a loss of AC power supply (Ref. 3).

Based on Technical Specifications Bases Section B.3.8.4, the 130 VDC battery minimum design voltage limit is 105 VDC. (Ref. 4).

This EAL is the hot condition equivalent of the cold condition loss of DC power EAL CU1.2.

Generic

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

Fermi Basis Reference(s):

1. Design Basis Document RXX-00 ESS Electrical AC Systems
2. UFSAR Chapter 8 Electrical Power
3. Design Bases Document R32-00 DC Electrical System
4. Technical Specifications Bases Section B.3.8.4 DC Sources - Operating
5. NEI 99-01 SG8

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**
Degraded voltage (< 105 VDC) on **both** 130 VDC system vital buses for ≥ 15 min. (Note 1)	

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

Mode Applicability:

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

Definition(s):

None

Basis:

Plant-Specific

At Fermi 2, the vital 260/130 VDC System ensures power is available for the reactor to be shutdown safely and maintained in a safe condition. The vital DC system is divided into two independent divisions - Division I and Division II - with separate DC power supplies. These power supplies consist of two separate 260/130V batteries and chargers serving systems such as RCIC, RHR, EDGs, and HPCI. The system provides sufficient capacity, via each of the Class 1E DC batteries, to power all required loads for 4 hours following a loss of AC power supply (Ref. 1).

Based on Technical Specifications Bases Section B.3.8.4, the 130 VDC battery minimum design voltage limit is 105 VDC. (Ref. 2).

Generic

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.

Fermi Basis Reference(s):

1. Design Bases Document R32-00 DC Electrical System
2. Technical Specifications Bases Section B.3.8.4 DC Sources - Operating
3. 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.

EAL:

SU3.1 Unusual Event
An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

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

Table S-2 Safety System Parameters
• Reactor power
• RPV water level
• RPV pressure
• Primary containment pressure
• Torus water level
• Torus temperature

Mode Applicability:

1 - Power Operation, 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.

Basis:

Plant-Specific

SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems.

The Primary Containment Monitoring System is an informational system that provides indications of the Primary Containment environmental variables such as Primary Containment Pressure and Suppression Pool level and temperature (Ref. 4)

The Integrated Plant Computer System (IPCS) is a computer system that provides the capability of monitoring, recording and displaying plant parameters via strategically located display devices. The IPCS is designed to be highly reliable and provide current information for selected plant variables. All realtime data displays update the current field conditions in a timely manner (Ref.1).

SPDS is a function of the IPCS that provides a specific selection of emergency response information. SPDS uses data from selected plant data systems and processes the data for display on the IPCS. SPDS information can be displayed on any IPCS terminal, which includes those specifically located in the control room. The SPDS display in the control room is provided to assist the operators in assessing the safety status of the plant following an accident (Ref. 1).

<u>Generic</u>

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.

Fermi Basis Reference(s):

1. UFSAR Section 7.6.1.9 Plant Computer Systems
2. AOP 20.615 Loss of Integrated Plant Computer System (IPCS)
3. ARP 3D17 IPCS Computer Trouble
4. SOP 23.408 Primary Containment Monitoring
5. NEI 99-01 SU2

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-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

Any Table S-3 transient event in progress

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

Table S-2 Safety System Parameters
• Reactor power • RPV water level • RPV pressure • Primary containment pressure • Torus water level • Torus temperature

Table S-3 Transient Events
• Automatic or manual runback > 25% thermal reactor power • Electrical load rejection > 25% full electrical load • Reactor scram • ECCS actuation • Thermal power oscillations > 10%

Mode Applicability:

1 - Power Operation, 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.

Basis:

Plant-Specific

SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems.

The Primary Containment Monitoring System is an informational system that provides indications of the Primary Containment environmental variables such as Primary Containment Pressure and Suppression Pool level and temperature (Ref. 4)

The Integrated Plant Computer System (IPCS) is a computer system that provides the capability of monitoring, recording and displaying plant parameters via strategically located display devices. The IPCS is designed to be highly reliable and provide current information for selected plant variables. All realtime data displays update the current field conditions in a timely manner (Ref.1).

SPDS is a function of the IPCS that provides a specific selection of emergency response information. SPDS uses data from selected plant data systems and processes the data for display on the IPCS. SPDS information can be displayed on any IPCS terminal, which includes those specifically located in the control room. The SPDS display in the control room is provided to assist the operators in assessing the safety status of the plant following an accident (Ref. 1).

Generic

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.

Fermi Basis Reference(s):

1. UFSAR Section 7.6.1.9 Plant Computer Systems
2. AOP 20.615 Loss of Integrated Plant Computer System (IPCS)
3. ARP 3D17 IPCS Computer Trouble

4. SOP 23.408 Primary Containment Monitoring
5. NEI 99-01 SA2

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 radiation monitor D11-K601A or D11-K601B ≥ high-high alarm (ARP 3D12) (Note 11)

Note 11: Consistent with Technical Specification 3.7.5, this EAL is applicable at all times while in Mode 1, Mode 2 or in Mode 3 with any main steam line not isolated and steam jet air ejector in operation.

Mode Applicability:

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

Definition(s):

None

Basis:

<u>Plant-Specific</u>

Elevated off-gas radiation activity is indicative of potential fuel clad failures and represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The Technical Specification allowable limit is 340 mCi/sec of noble gases measured at the discharge of the 2.2 minute delay piping (Ref.3). The high-high radiation alarm setpoint is set to alert operators that the Technical Specification release limit may be exceeded (Ref. 1, 2). The high-high radiation alarm setpoint has been conservatively selected because it is operationally significant and is readily recognizable by Control Room operating staff. Consistent with Technical Specification 3.7.5, EAL SU4.1 is applicable at all times while in Mode 1 (Power Operation) and in Mode 2 (Startup) or Mode 3 (Hot Shutdown) with any main steam line not isolated and steam jet air ejector (SJAE) in operation. (Ref. 1)

<u>Generic</u>

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.

Fermi Basis Reference(s):

1. ARP 3D12 DIV. 1/2 OFFGAS RADN MONITOR HIGH/HIGH
2. AOP 20.000.07 Fuel Cladding Failure
3. Technical Specifications Section 3.7.5 Main Condenser Offgas
4. 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**
Sample analysis indicates that a reactor coolant activity value is > an allowable limit specified in Technical Specifications (Note 12)	

Note 12: Consistent with Technical Specification 3.4.7, this EAL is applicable at all times while in Mode 1, Mode 2 or in Mode 3 with any main steam line not isolated.

Mode Applicability:

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

Definition(s):

None

Basis:

Plant-Specific

This EAL addresses RCS specific activity exceeding the limits of Technical Specifications Section 3.4.7, which are: (1) 0.2 μCi per gram DEI-131 for more than 48 hours, or (2) 4.0 μCi per gram DEI-131. Consistent with Technical Specification 3.4.7, EAL SU4.2 is applicable at all times while in Mode 1 (Power Operation) and in Mode 2 (Startup) or Mode 3 (Hot Shutdown) with any main steam line not isolated (Ref. 1).

Generic

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.

Fermi Basis Reference(s):

1. Technical Specifications 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 Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

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

Definition(s):

None

Basis:

Plant-Specific

Unidentified leakage and identified leakage are determined by performance of the RCS water inventory balance (IPCS CHG_{NET}, LRATE). Pressure boundary leakage would first appear as unidentified leakage and can only be positively identified by inspection (Ref. 1).

Technical Specifications defines RCS leakage as follows (Ref. 1, 2):

Identified leakage:

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. 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 LEAKAGE.

Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE.

Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

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. 3).

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.

Generic

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.

Fermi Basis Reference(s):

1. Technical Specifications Section 1.1 Definitions - Leakage
2. Technical Specifications Section 3.4.4 RCS Operational Leakage
3. UFSAR Chapter 5 Reactor Coolant System and Connected Systems Section 5.1 Summary Description
4. NEI 99-01 SU4

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 COP H11-P603 is successful in shutting down the reactor as indicated by reactor power < 3% (Note 8, 9).

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.

Note 9: For manual scram actions, the reactor scram pushbuttons, taking the Reactor Mode Switch to Shutdown or manual initiation of ARI on COP H11-P603 are the only methods applicable to this EAL.

Mode Applicability:

1 - Power Operation, 2 - Startup

Definition(s):

None

Basis:

Plant-Specific

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 trip setpoint of 3% (Ref. 1).

For the purposes of this EAL, a successful automatic initiation of ARI that reduces reactor power below 3% is not 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.

For purposes of emergency classification, a "successful" manual reactor scram includes only those actions taken by the reactor operator in the Control Room on the reactor control console (COP H11-P603). These actions include the manual scram pushbuttons, placing the Reactor Mode Switch in Shutdown and manual initiation of ARI. These pushbuttons and controls can be rapidly manipulated from the Control Room panels.

If the above described response cannot be verified, operators perform contingency actions that manually insert control rods or implement alternate control rod insertion methodologies performed either away from the reactor control console or external to the Control Room. Those actions required to be performed away from the reactor control console or outside of the Control Room to initiate rapid control rod insertion are not considered a "successful" manual reactor scram.

In the event that the operator identifies a reactor scram is imminent and successfully initiates a manual reactor scram before the automatic trip 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 in the Control Room fail to reduce reactor power below the power associated with the safety system design (< 3%) following a failure of an automatic scram, the event escalates to an Alert under EAL SA6.1.

The APRM downscale trip setpoint (3%) is a minimum reading on the power range scale that indicates power production (Ref. 1). At or below the APRM downscale trip setpoint,

plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM/IRM) indications or other reactor parameters (e.g., number of open SRVs, number of open main turbine bypass valves, main steam flow, RPV pressure and torus water temperature trend, etc.) can be used to determine if reactor power is greater than or equal to 3% power.

By definition, an operating mode change occurs when the Mode Switch is moved from the startup or run position to the shutdown position. The plant operating mode that existed at the time the event occurs (i.e., Power Operation or Startup), however, requires emergency classification of at least an Unusual Event. The operating mode change associated with movement of the Mode Switch, by itself, does not justify failure to declare an emergency for ATWS events.

<u>Generic</u>

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.

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.

Fermi Basis Reference(s):

1. EOP Support Documentation Section 1 Plant Specific Technical Guideline
2. AOP 20.000.21 Reactor Scram
3. 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 COP H11-P603 is successful in shutting down the reactor as indicated by reactor power < 3% (Note 8, 9).

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.

Note 9: For manual scram actions, the reactor scram pushbuttons, taking the Reactor Mode Switch to Shutdown or manual initiation of ARI on COP H11-P603 are the only methods applicable to this EAL.

Mode Applicability:

1 - Power Operation, 2 - Startup

Definition(s):

None

Basis:

Plant-Specific

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 < 3%).

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 trip setpoint of 3% (Ref. 1).

For the purposes of this EAL, a successful automatic initiation of ARI that reduces reactor power to or below 3% is not 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.

For purposes of emergency classification, a "successful" manual reactor scram includes only those actions taken by the reactor operator in the Control Room on the reactor control console (COP H11-P603). These actions include the manual scram pushbuttons, placing the Reactor Mode Switch in Shutdown and manual initiation of ARI. These pushbuttons and controls can be rapidly manipulated from the Control Room panels.

Taking the Mode Switch out of Run position reduces the APRM scram setpoint to 15% reactor power. If reactor power is ≥ 15% at the time the Mode Switch is taken out of the Run position, an automatic RPS scram signal will have been generated in addition to the manual scram signal generated by taking the Mode Switch to Shutdown. Should the other immediate manual scrams not be successful in reducing reactor power to < 3%, an Alert should be declared based on exceeding EAL SA6.1.

If the above described response cannot be verified, operators perform contingency actions that manually insert control rods or implement alternate control rod insertion methodologies performed either away from the reactor control console or external to the Control Room. Those actions required to be performed away from the reactor control console or outside of the Control Room to initiate rapid control rod insertion are not considered a "successful" manual reactor scram.

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 (< 3%) following a failure of an initial manual scram, the event escalates to an Alert under EAL SA6.1.

The APRM downscale trip setpoint (3%) is a minimum reading on the power range scale that indicates power production (Ref. 1). At or below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM/IRM) indications or other reactor parameters (e.g., number of open SRVs, number of open main turbine bypass valves, main steam flow, RPV pressure and torus water temperature trend, etc.) can be used to determine if reactor power is greater than or equal to 3% power.

By definition, an operating mode change occurs when the Mode Switch is moved from the startup or run position to the shutdown position. The plant operating mode that existed at the time the event occurs (i.e., Power Operation or Startup), however, requires emergency classification of at least an Unusual Event. The operating mode change associated with movement of the Mode Switch, by itself, does not justify failure to declare an emergency for ATWS events.

<u>Generic</u>

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.

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.

Fermi Basis Reference(s):

1. EOP Support Documentation Section 1 Plant Specific Technical Guideline
2. AOP 20.000.21 Reactor Scram
3. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – 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 COP H11-P603 are **not** successful in shutting down the reactor as indicated by reactor power $\geq$ 3% (Note 8, 9)

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.

Note 9: For manual scram actions, the reactor scram pushbuttons, taking the Reactor Mode Switch to Shutdown or manual initiation of ARI on COP H11-P603 are the only methods applicable to this EAL.

Mode Applicability:

1 - Power Operation, 2 - Startup

Definition(s):

None

Basis:

Plant-Specific

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 ($\geq$ 3%) (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 trip setpoint of 3% (Ref. 1).

For the purposes of this EAL, a successful automatic initiation of ARI that reduces reactor power below 3% is not 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.

For purposes of emergency classification, a “successful” manual reactor scram includes only those actions taken by the reactor operator in the Control Room on the reactor control console (COP H11-P603). These actions include the manual scram pushbuttons, placing the Reactor Mode Switch in Shutdown and manual initiation of ARI. These pushbuttons and controls can be rapidly manipulated from the Control Room panels.

If the above described response cannot be verified, operators perform contingency actions that manually insert control rods or implement alternate control rod insertion methodologies performed either away from the reactor control console or external to the Control Room. Those actions required to be performed away from the reactor control console or outside of the Control Room to initiate rapid control rod insertion are not considered a “successful” manual reactor scram.

If subsequent manual reactor scram actions in the Control Room fail to reduce reactor power below the power associated with the safety system design (< 3%) following a failure of an initial automatic or manual scram, the event is classifiable under this EAL.

The APRM downscale trip setpoint (3%) is a minimum reading on the power range scale that indicates power production (Ref. 1). At or below the APRM downscale trip setpoint,

plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM/IRM) indications or other reactor parameters (e.g., number of open SRVs, number of open main turbine bypass valves, main steam flow, RPV pressure and torus water temperature trend, etc.) can be used to determine if reactor power is greater than 3% power.

By definition, an operating mode change occurs when the Mode Switch is moved from the startup or run position to the shutdown position. The plant operating mode that existed at the time the event occurs (i.e., Power Operation or Startup), however, requires emergency classification of at least an Unusual Event. The operating mode change associated with movement of the Mode Switch, by itself, does not justify failure to declare an emergency for ATWS events.

Generic

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 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 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.

Fermi Basis Reference(s):

1. EOP Support Documentation Section 1 Plant Specific Technical Guideline
2. AOP 20.000.21 Reactor Scram
3. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 6 – 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 $\geq$ 3%

AND

EITHER of the following conditions exist:

- RPV water level **cannot** be restored and maintained > -25 in.
- Torus water temperature and RPV pressure **cannot** be maintained below the Heat Capacity Limit (HCL)

Mode Applicability:

1 - Power Operation, 2 - Startup

Definition(s):

None

Basis:

Plant-Specific

This EAL addresses the following:

- Any automatic or manual reactor scram signal followed by a failure of all subsequent methods to shut down the reactor, both within and external to the Control Room, to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed ($\geq$ 3%, Ref. 1, 2), and
- Indications that either core cooling is extremely challenged (RPV water level cannot be restored and maintained above the Minimum Steam Cooling RPV Water Level) or heat removal is extremely challenged (torus water temperature and RPV pressure cannot be maintained below the Heat Capacity Limit) (Ref. 1).

For this Site Area Emergency EAL, reactor shutdown achieved by injection of boron or use of the alternate control rod insertion methods of 29.ESP.03 is also credited 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. 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 RPV water level cannot be restored and maintained above -25 in. (Ref. 1,4). The Minimum Steam Cooling RPV Water Level (MSCRWL) is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. Consistent with the EOP definition of “cannot be restored and maintained,” the determination that RPV water level cannot be restored and maintained above the MSCRWL may be made at, before, or after RPV water level actually decreases to this point.

Indication that core heat removal is extremely challenged is manifested by the inability to maintain torus water temperature and RPV pressure below the Heat Capacity Limit (HCL). The HCL is the highest torus water temperature from which emergency RPV depressurization will not raise (Ref. 1, 5):

- Torus water temperature above the design value, or
- Torus pressure above Primary Containment Pressure Limit before the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

The HCL is a function of RPV pressure and torus water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant. Plant parameters in excess of the HCL could be a precursor of Primary Containment failure. (Ref. 1, 5, 6)

The HCL is given in EOP 29.100.01 Sheet 6 Curves, Cautions and Tables (Ref. 5). This threshold is met when Emergency RPV Depressurization is required in EOP Primary

Containment Control, Step TWT-5 (Ref. 6). This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature.

Generic

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 shutdown the reactor.

Escalation of the emergency classification level would be via IC RG1 or FG1.

Fermi Basis Reference(s):

1. EOP Support Documentation Section 1 Plant Specific Technical Guideline
2. AOP 20.000.21 Reactor Scram
3. 29.ESP.03 Alternate Control Rod Insertion Methods
4. EOP 29.100.01 Sheet 1A RPV Control - ATWS
5. EOP 29.100.01 Sheet 6 Curves, Cautions and Tables
6. EOP 29.100.01 Sheet 2 Primary Containment Control
7. NEI 99-01 SS5

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-4 onsite communication methods

OR

Loss of **all** Table S-4 offsite communication methods

OR

Loss of **all** Table S-4 NRC communication methods

Table S-4 Communication Methods			
System	**Onsite**	**Offsite**	**NRC**
Administrative Telephones	X	X	X
RERP Emergency Telephones	X	X	X
Satellite Phones		X	X
Federal Telephone System (ENS)		X	X
Automatic Ring Lines		X	
MI State Radios (800 MHz)		X	
Plant Radio System	X		
Hi-Com (PA System)	X		

Mode Applicability:

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

Definition(s):

None

Basis:

Plant-Specific

The Table S-4 list for onsite communications loss encompasses the loss of all means of routine communications (e.g., administrative and internal telephones, plant page [Hi-Com] and plant radios) (Ref. 1, 2).

The Table S-4 list for offsite communications loss encompasses the loss of all means of communications with offsite authorities. This includes the RERP telephone dedicated ring lines, backup phone systems administrative telephone lines, satellite, and FTS (ENS) which can be utilized as a regular telephone (Ref. 1, 2).

The Table S-4 list for NRC communications loss encompasses the loss of all means of communications with the NRC. This includes the FTS (ENS), backup phone systems (administrative telephone lines, RERP phones and satellite) (Ref. 1, 2).

The communications methods used at Fermi 2 are described in the RERP Plan (Ref. 1).

The radio network at Fermi 2 involves several radio systems to effect communications within the plant with damage control teams, rescue teams, fire brigade, radiological monitoring teams, and security personnel as well as provide backup communications to essential Offsite Emergency Response Organizations (OROs) in the event of telephone equipment malfunction.

There are two radio consoles normally used in the Control Room. One is installed in panel H11-P700 to establish communications using plant radio zone 1 (control room group) to hand-held portable radios (OPS channel 1 or 2) via the plant radio repeater system.

An additional radio console is located in panel H11-P703 to allow for backup communications to hand-held portable radios on various other user groups via plant radio zone 1 repeater system or backup repeaters (zone 2). Maintenance channels 1, 2, or 3 can also be selected at this station. This console also provides a backup radio communication selection into security zone 3 that provides another two repeaters for radio operation.

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

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

<u>Generic</u>

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

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

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 State, Monroe and Wayne County EOCs

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

Fermi Basis Reference(s):

1. Fermi Emergency Plan Section F Emergency Communications
2. EP-580 Equipment Important to Emergency Response
4. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 8 – Hazardous Event Affecting Safety Systems

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

EAL:

SA8.1 Alert

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

AND

EITHER of the following:

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

Table S-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:

1 - Power Operation, 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.

Basis:

<u>Plant-Specific</u>

- The significance of seismic events are discussed under EAL HU2.1 (Ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (Ref. 2, 3).
- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 90 mph. (Ref. 4).

- Areas containing functions and systems required for safe shutdown of the plant are identified by Fire Zone in the fire response procedure (Ref. 5).
- An explosion (including a steam line 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. The need to classify a steam line break not considered an explosion itself is considered in fission product barrier degradation monitoring (EAL Category F).

Generic

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.

Fermi Basis Reference(s):

1. AOP 20.000.01 Acts of Nature
2. AOP 20.000.03 Turbine Building Flooding
3. PLG-0849 Fermi 2 Internal Flooding Analysis
4. UFSAR Section 3.3.3.1 Design Wind Speed
5. AOP 20.000.22 Plant Fires
6. NEI 99-01 SA9

<u>Category F – Fission Product Barrier Degradation</u>

EAL Group: Hot Conditions (RCS temperature > 200ºF); 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. <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 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 suppression pool, 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 ECL from Alert to 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. 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 number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

<u>Alert:</u> ***Any*** *loss or* ***any*** *potential loss of either Fuel Clad or RCS*

<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*

The logic used for Category F EALs reflects 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 fission product barriers are addressed in Recognition Category S.

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 plant-specific Fermi 2 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 to be 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, then 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 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: **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 barrier (Table F-1)	

Mode Applicability:

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

Definition(s):

None

Basis:

Plant-Specific

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 Primary Containment barrier. Unlike the Primary Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Primary Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1.

Generic

None

Fermi Basis Reference(s):

1. NEI 99-01 FA1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss or potential loss of **any** two barriers

EAL:

FS1.1	**Site Area Emergency**
Loss or potential loss of **any** two barriers (Table F-1)	

Mode Applicability:

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

Definition(s):

None

Basis:

Plant-Specific

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 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 Primary Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss

thresholds existed, the Emergency Director would have greater assurance that escalation to a General Emergency is less imminent.

<u>Generic</u>

None

Fermi Basis Reference(s):

1. NEI 99-01 FS1

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 Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

Plant-Specific

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 General Emergency classification level each barrier is weighted equally. A General Emergency is therefore appropriate for any combination of the following conditions:

- Loss of Fuel Clad, RCS and Primary Containment barriers
- Loss of Fuel Clad and RCS barriers with potential loss of Primary Containment barrier
- Loss of RCS and Primary Containment barriers with potential loss of Fuel Clad barrier
- Loss of Fuel Clad and Primary Containment barriers with potential loss of RCS barrier

Generic

None

Fermi Basis Reference(s):

1. NEI 99-01 FG1

Category E – Independent Spent Fuel Storage Installation (ISFSI)

EAL Group: ALL (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.

A Notification of 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.

Category: ISFSI

Subcategory: Confinement Boundary

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Unusual Event

Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than **EITHER of** the following on the surface of the spent fuel cask (overpack):

- 60 mrem/hr (γ + n) on the top of the overpack

OR

- 600 mrem/hr (γ + n) on the side of the overpack excluding inlet and outlet ducts

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 applied to the Fermi 2 ISFSI, the CONFINEMENT BOUNDARY is defined to be the HI-STORM Multi-Purpose Canister (MPC).

Basis:

Plant-Specific

Overpacks are the casks which receive and contain the sealed Multi-Purpose Canisters (MPCs) for interim storage on the ISFSI. They provide gamma and neutron shielding, and provide for ventilated air flow to promote heat transfer from the MPC to the environs. The term overpack does not include the transfer cask (Ref. 1).

MPCs are the sealed spent nuclear fuel canisters which consist of a honeycombed fuel basket contained in a cylindrical canister shell which is welded to a baseplate, lid with welded port cover plates, and closure ring. The MPC provides the CONFINEMENT BOUNDARY for the contained radioactive materials (Ref. 1).

The values shown represent 2 times the limits specified in the ISFSI Certificate of Compliance Technical Specification section 5.7.4 for radiation external to a loaded MPC overpack (Ref. 1).

Generic

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", which is also used in Recognition Category R IC RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

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

Fermi Basis Reference(s):

1. Certificate of Compliance No. 1014 Appendix A Technical Specifications for the HI-STORM 100 Cask System

ATTACHMENT 2

FISSION PRODUCT BARRIER 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 Primary Containment). The table is structured so that the three barriers occupy adjacent columns. Each fission product barrier column is further divided into two columns; one for Loss thresholds and one for Potential Loss thresholds.

The first column of the table (to the left of the Fuel Clad Barrier Loss column) lists the categories (types) of fission product barrier thresholds. The fission product barrier categories are:

1. RPV Water Level
2. RCS Leak Rate
3. Primary Containment Conditions
4. Primary Containment Radiation / RCS Activity
5. Primary Containment Integrity or Bypass
6. Emergency Director Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the category rows and the Loss/Potential Loss columns. The intersection of each category 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 letters within each Loss and Potential Loss column beginning with “A.” In this manner, a threshold can be identified by its category number and threshold letter. For example, the first Fuel Clad barrier Loss in Category 2 is “FC Loss 2.A,” the third Primary Containment barrier Potential Loss in Category 4 is “PC P-Loss 4.C,” etc.

If a cell in Table F-1 contains more than one threshold, each of the 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 row of fission product barrier Loss and Potential Loss thresholds in that category to determine if any threshold has been exceeded. If a threshold has not been exceeded in that category row, the EAL-user proceeds to the next likely category and continues review of the row of thresholds in the new category.

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 Primary Containment radiation is sufficiently high (i.e., > 10,000 R/hr), a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Primary Containment barrier exist. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, FA1.1 and FU1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad barrier threshold bases appear first, followed by the RCS barrier and finally the Primary Containment barrier threshold bases. In each barrier, the bases are given according to category Loss followed by category Potential Loss beginning with Category 1, then 2…5.

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad Barrier		Reactor Coolant System Barrier		Primary Containment Barrier	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
1 **RPV Water Level**	A. Inadequate core cooling as indicated by **ANY** of the following: 1. RPV water level **cannot** be restored and maintained > -48 in. with ≥ 5725 gpm Core Spray loop flow **OR** 2. RPV water level **cannot** be restored and maintained > -25 in with < 5725 gpm Core Spray loop flow **OR** 3. RPV water level **cannot** be determined and core damage is occurring	A. RPV water level cannot be restored and maintained above 0 in. (TAF) or cannot be determined	A. RPV water level cannot be restored and maintained above 0 in. (TAF) or cannot be determined	None	None	A. Inadequate core cooling as indicated by **ANY** of the following: 1. RPV water level **cannot** be restored and maintained > -48 in. with ≥ 5725 gpm Core Spray loop flow **OR** 2. RPV water level **cannot** be restored and maintained > -25 in with < 5725 gpm Core Spray loop flow **OR** 3. RPV water level **cannot** be determined and core damage is occurring
2 **RCS Leak Rate**	None	None	A. UNISOLABLE break in **any** of the following: • Main Steam Line • HPCI Steam Line • RCIC Steam Line • RWCU • Feedwater **OR** B. Emergency RPV Depressurization is required	A. UNISOLABLE primary system leakage into Secondary Containment that results in exceeding **EITHER** of the following: 1. One or more Secondary Containment Control **Max Normal** Operating Temperatures (EOP Table 12) **OR** 2. One or more Secondary Containment Control **Max Normal** Operating Area Radiation Levels (EOP Table 14)	A. UNISOLABLE primary system leakage into Secondary Containment that results in exceeding **EITHER** of the following: 1. One or more Secondary Containment Control **Max Safe** Operating Temperatures (EOP Table 12) **OR** 2. Exceeding Secondary Containment Control **Max Safe** Operating Area Radiation Level on channel 14 ARM (RBSB Torus Room)	None
3 **PC Conditions**	None	None	A. Drywell pressure > 1.68 psig due to RCS leakage	None	A. UNPLANNED rapid drop in Primary Containment pressure following Primary Containment pressure rise **OR** B. Primary Containment pressure response not consistent with LOCA conditions	A. Primary Containment Pressure > 62 psig **OR** B. > 6% H_2 AND > 5% O_2 in **EITHER** the drywell or suppression chamber **OR** C. EOP Heat Capacity Limit (HCL) exceeded

Table F-1 Fission Product Barrier Threshold Matrix						
	Fuel Clad Barrier		Reactor Coolant System Barrier		Primary Containment Barrier	
Category	Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
4 **PC Rad / RCS Activity**	A. CHRRM reading > 2.25E+3 R/hr **OR** B. Primary coolant activity > 300 µCi/gm DEI-131	None	A. CHRRM reading > 8.72E+1 R/hr	None	None	A. CHRRM reading > 1.79E+4 R/hr
5 **PC Integrity or Bypass**	None	None	None	None	A. UNISOLABLE direct downstream pathway to the environment exists after Primary Containment isolation signal **OR** B. Intentional Primary Containment venting, irrespective of offsite radioactivity release rates, per EOPs	None
6 **ED Judgment**	A. **Any** condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier	A. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier	A. **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	A. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	A. **Any** condition in the opinion of the Emergency Director that indicates loss of the Primary Containment barrier	A. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the Primary Containment barrier

Barrier: Fuel Clad

Category: 1. RPV Water Level

Degradation Threat: Loss

Threshold:

A. Inadequate core cooling as indicated by <u>ANY</u> of the following:

1. RPV water level **cannot** be restored and maintained > -48 in. with ≥ 5725 gpm Core Spray loop flow

   OR

2. RPV water level **cannot** be restored and maintained > -25 in. with < 5725 gpm Core Spray loop flow

   OR

3. RPV water level **cannot** be determined and core damage is occurring

Definition(s):

None

Basis:

<u>Plant-Specific</u>

Requirements for entry into the Severe Accident Guidelines (SAGs) are established in EOP RPV Control, EOP RPV Control - ATWS and EOP RPV Flooding & EOP RPV Flooding - ATWS (Ref. 1, 2, 3, 4, 5). 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.

Direction is specified that SAG entry is required when:

- EOP RPV Control - RPV water level cannot be restored and maintained above -48 in. with required Core Spray flow.
- EOP RPV Control - RPV water level cannot be restored and maintained above -25 in. (MSCRWL) with insufficient Core Spray flow.
- EOP RPV Control - ATWS - RPV water level cannot be restored and maintained above -25 in. (MSCRWL).

- EOP Flooding & EOP Flooding - ATWS - RPV water level cannot be determined and it is determined that core damage is occurring.

This threshold is also a Potential Loss of the Containment barrier (PC P-Loss 1.A). Since SAG entry occurs after core uncovery has occurred, a Loss of the RCS barrier exists (RCS Loss 1.A). SAG entry, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

Generic

The Loss threshold represents the EOP requirement for SAG entry.

Fermi Basis Reference(s):

1. EOP 29.100.01 Sheet 1 RPV Control
2. EOP 29.100.01A Sheet 1 RPV Control - ATWS
3. EOP 29.100.01 Sheet 3 RPV Flooding, Emerg. Depress. & Steam Cooling
4. EOP 29.100.01 Sheet 3A RPV Flooding, Emerg. Depress. & Steam Cooling - ATWS
5. EOP Support Documentation Section 1 Plant Specific Technical Guideline
6. NEI 99-01 RPV Water Level Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: 1. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

A. RPV water level cannot be restored and maintained above 0 in. (TAF) or cannot be determined

Definition(s):

None

Basis:

Plant-Specific

An RPV water level instrument reading of 0 in. indicates RPV water level is at the top of active fuel. When RPV water level is at or above the top of active fuel, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV water level is below the top of active fuel following depressurization of the RPV (automatically, manually or by failure of the RCS barrier), the uncovered portion of the core must be cooled by less reliable means (i.e., 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 (Ref. 1).

This threshold is considered to be exceeded when, as specified in the 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. Consistent with the EOP definition of “cannot be restored and maintained,” the determination that RPV water level cannot be restored and maintained above the top of active fuel may be made at, before, or after RPV water level actually decreases to this point. (Ref. 1)

When RPV water level cannot be determined, EOPs require RPV flooding strategies. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP RPV Flooding and EOP RPV Flooding - ATWS 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 ATWS events). (Ref. 2, 3). If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the fuel clad barrier exists.

Note that EOP RPV Control - ATWS may require intentional uncovery of the core and control of RPV water level between 0 in. and -25 in., the Minimum Steam Cooling RPV Water Level (MSCRWL) (Ref. 4). Under these conditions, a ATWS greater then design decay heat level event exists and requires at least an Alert classification in accordance with the RPS Failure EALs.

<u>Generic</u>

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 1.A. 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.

Fermi Basis Reference(s):

1. EOP Support Documentation Section 1 Plant Specific Technical Guideline
2. EOP 29.100.01 Sheet 3 RPV Flooding, Emerg. Depress. & Steam Cooling
3. EOP 29.100.01 Sheet 3 RPV Flooding, Emerg. Depress. & Steam Cooling - AWS
4. EOP 29.100.01A Sheet 1 RPV Control - ATWS
5. NEI 99-01 RPV Water Level Fuel Clad Potential Loss 2.A

Barrier: Fuel Clad

Category: 2. RCS Leak Rate

Degradation Threat: Loss

Threshold:

None

Definition(s):

None

Basis:

Plant-Specific

N/A

Generic

N/A

Fermi Basis Reference(s):

N/A

Barrier: Fuel Clad

Category: 2. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Definition(s):

None

Basis:

N/A

Generic

N/A

Fermi Basis Reference(s):

N/A

Barrier: Fuel Clad

Category: 3. PC Conditions

Degradation Threat: Loss

Threshold:

None

Definition(s):

None

Basis:

Plant-Specific

N/A

Generic

N/A

Fermi Basis Reference(s):

N/A

Barrier: Fuel Clad

Category: 3. PC Conditions

Degradation Threat: Potential Loss

Threshold:

None

Definition(s):

None

Basis:

Plant-Specific

N/A

Generic

N/A

Fermi Basis Reference(s):

N/A

Barrier: Fuel Clad

Category: 4. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

A. CHRRM reading > 2.25E+3 R/hr

Definition(s):

None

Basis:

Plant-Specific

For Fermi 2, the Containment High Range Radiation Monitor (CHRRM) is used to measure drywell radiation levels. A valid CHRRM reading of 2.25E+3 R/hr corresponds to 2.5% gap release (300 µCi/gm DEI I-131) discharged instantaneously into containment atmosphere (Ref. 1).

Generic

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 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold 4.A since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the emergency classification level to a Site Area Emergency.

There is no Potential Loss threshold associated with Primary Containment Radiation.

Fermi Basis Reference(s):

1. EP-EALCALC-FERMI-1402 Fermi EAL Technical Bases Calculations – CHRRM Series Rev. 0
2. NEI 99-01 Primary Containment Radiation Fuel Clad Loss 4.A

Barrier: Fuel Clad

Category: 4. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

B. Primary coolant activity > 300 µCi/gm DEI-131

Definition(s):

None

Basis:

Plant-Specific

300 µCi/gm DEI-131 is equivalent to 2.5% fuel clad (gap) damage (Ref. 1).

Generic

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

There is no Potential Loss threshold associated with RCS Activity.

There is no Potential Loss threshold associated with Primary Containment Radiation.

Fermi Basis Reference(s):

1. EP-EALCALC-FERMI-1402 Fermi EAL Technical Bases Calculations – CHRRM Series Rev. 0
2. NEI 99-01 RCS Activity Fuel Clad Loss 1.A

Barrier: Fuel Clad

Category: 4. PC Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Definition(s):

None

Basis:

<u>Plant-Specific</u>

N/A

<u>Generic</u>

N/A

Fermi Basis Reference(s):

N/A

Barrier: Fuel Clad

Category: 5. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Definition(s):

None

Basis:

Plant-Specific

N/A

Generic

N/A

Fermi Basis Reference(s):

N/A

Barrier: Fuel Clad

Category: 5. PC Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Definition(s):

None

Basis:

Plant-Specific

N/A

Generic

N/A

Fermi Basis Reference(s):

N/A

Barrier: Fuel Clad

Category: 6. ED Judgment

Degradation Threat: Loss

Threshold:

A. **ANY** condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier

Definition(s):

None

Basis:

Plant-Specific

None

Generic

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

Fermi Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Fuel Clad Loss 6.A

Barrier: Fuel Clad

Category: 6. ED Judgment

Degradation Threat: Potential Loss

Threshold:

A. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier

Definition(s):

None

Basis:

Plant-Specific

None

Generic

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

Fermi Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Potential Fuel Clad Loss 6.A

Barrier: Reactor Coolant System

Category: 1. RPV Water Level

Degradation Threat: Loss

Threshold:

A. RPV water level cannot be restored and maintained above 0 in. (TAF) or cannot be determined.

Definition(s):

None

Basis:

Plant-Specific

An RPV water level instrument reading of 0 in. indicates RPV water level is at the top of active fuel. When RPV water level is at or above the top of active fuel, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV water level is below the top of active fuel following depressurization of the RPV (automatically, manually or by failure of the RCS barrier), the uncovered portion of the core must be cooled by less reliable means (i.e., 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 (Ref. 1).

Consistent with the EOP definition of "cannot be restored and maintained," the determination that RPV water level cannot be restored and maintained above the top of active fuel may be made at, before, or after RPV water level actually decreases to this point. (Ref. 1)

When RPV water level cannot be determined, EOPs require RPV flooding strategies. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP RPV Flooding and EOP RPV Flooding - ATWS 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 ATWS events). (Ref. 2, 3). If RPV water level cannot be restored and maintained above the top of active fuel or RPV water level cannot be determined with respect to the top of active fuel, a loss of the RCS barrier exists.

Note that EOP RPV Control - ATWS may require intentional uncovery of the core and control of RPV water level between 0 in. and -25 in., the Minimum Steam Cooling RPV Water Level (MSCRWL) (Ref. 4). Under these conditions, an ATWS above design decay heat level event exists and requires at least an Alert classification in accordance with the RPS Failure EALs.

Generic

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 1.A. 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.

Fermi Basis Reference(s):

1. EOP Support Documentation Section 1 Plant Specific Technical Guideline
2. EOP 29.100.01 Sheet 3 RPV Flooding, Emerg. Depress. & Steam Cooling
3. EOP 29.100.01 Sheet 3 RPV Flooding, Emerg. Depress, & Steam Cooling - AWS
4. EOP 29.100.01A Sheet 1 RPV Control - ATWS
5. NEI 99-01 RPV Water Level RCS Loss 2.A

Barrier: Reactor Coolant System

Category: 1. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

None

Definition(s):

None

Basis:

Plant-Specific

N/A

Generic

N/A

Fermi Basis Reference(s):

N/A

Barrier: Reactor Coolant System

Category: 2. RCS Leak Rate

Degradation Threat: Loss

Threshold:

A. 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.

Basis:

Plant-Specific

The list of systems included in this threshold are the high energy lines which, if ruptured and remain unisolated, can rapidly depressurize the RPV. These lines are typically isolated by actuation of the Leak Detection system.

Large high-energy line breaks such as Main Steam Line (MSL), High Pressure Coolant Injection (HPCI), Feedwater (failure of non-return valves), Reactor Water Cleanup (RWCU) or Reactor Core Isolation Cooling (RCIC) that are UNISOLABLE represent a significant loss of the RCS barrier. Determination of whether the leak is isolated to preclude EAL declaration must occur within the 15-minute assessment period. (Ref.1)

The conditions of this threshold include required containment isolation failures allowing a flow path to the environment. A release pathway outside Primary Containment exists when flow is not prevented by downstream isolations. Emergency declaration under this threshold would not be required in the case of a failure of both isolation valves to close but no downstream flowpath exists. 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 5.A) barriers and justifies declaration of a Site Area Emergency (i.e., Loss or Potential Loss of any two barriers).

Generic

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 from the Control Room, the RCS barrier Loss threshold is met.

Fermi Basis Reference(s):

1. UFSAR Section 6.2.4.2.2.2.2 Effluent Lines
2. NEI 99-01 RCS Leak Rate RCS Loss 3.A

Barrier: Reactor Coolant System

Category: 2. RCS Leak Rate

Degradation Threat: Loss

Threshold:

B. Emergency RPV Depressurization is required

Definition(s):

None

Basis:

Plant-Specific

Emergency RPV Depressurization is specified in the EOP flowcharts (EOP Emergency Depressurization) when symbols containing the phrase “EMERGENCY RPV DEPRESS IS REQ'D” are reached. (Ref. 1, 2).

Generic

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.

Fermi Basis Reference(s):

1. EOP Support Documentation Section 1 Plant Specific Technical Guideline
2. EOP 29.100.01 Sheet 3 RPV Flooding, Emerg. Depress, & Steam Cooling
3. NEI 99-01 RCS Leak Rate RCS Loss 3.B

Barrier: Reactor Coolant System

Category: 2. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

A. UNISOLABLE primary system leakage into Secondary Containment that results in exceeding **EITHER** of the following:
   1. One or more Secondary Containment Control **Max Normal** Operating Temperatures (EOP Table 12)
      OR
   2. One or more Secondary Containment Control **Max Normal** Operating Area Radiation Levels (EOP Table 14)

Definition(s):

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

Basis:

Plant-Specific

The presence of elevated general area temperatures or radiation levels in the Reactor Building (RB) may be indicative of UNISOLABLE primary system leakage outside the Primary Containment. When parameters reach the threshold 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. Determination of whether the leak is isolated to preclude EAL declaration must occur within the 15-minute assessment period. (Ref. 1, 2)

In general, multiple indications should be used to determine if a primary system is discharging outside Primary Containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the secondary containment since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room flooding, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or

unexpected main turbine control valve closure) may indicate that a primary system is discharging into the secondary containment.

<u>Generic</u>

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 which is indicated by Max Normal Operating values escalates to a Site Area Emergency when combined with Containment Barrier Loss threshold 3.A (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

Fermi Basis Reference(s):

1. EOP Support Documentation Section 1 Plant Specific Technical Guideline
2. EOP 29.100.01 Sheet 5 Secondary Containment and Rad Release
3. NEI 99-01 RCS Leak Rate RCS Potential Loss 3.A

Barrier: Reactor Coolant System

Category: 3. PC Conditions

Degradation Threat: Loss

Threshold:

A. Drywell pressure > 1.68 psig due to RCS leakage

Definition(s):

None

Basis:

Plant-Specific

The drywell high pressure scram setpoint is an entry condition to the EOP flowcharts: EOP RPV Control, and EOP Primary Containment Control (Ref. 1, 2, 3). Normal Primary Containment (PC) pressure control functions such as operation of drywell cooling and venting are specified in EOP Primary Containment Control in advance of less desirable but more effective functions such as operation of drywell or torus sprays.

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. 1).

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. Primary Containment pressure greater than 1.68 psig with corollary indications (e.g., elevated drywell temperature, indications of loss of RCS inventory) should, therefore, be considered a Loss of the RCS barrier.

Generic

1.68 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.

Fermi Basis Reference(s):

1. EOP Support Documentation Section 1 Plant Specific Technical Guideline
2. EOP 29.100.01 Sheet 1 RPV Control
3. EOP 29.100.01 Sheet 2 Primary Containment Control
4. NEI 99-01 Primary Containment Pressure RCS Loss 1.A

Barrier: Reactor Coolant System

Category: 3. PC Conditions

Degradation Threat: Potential Loss

Threshold:

None

Definition(s):

None

Basis:

Plant-Specific

N/A

Generic

N/A

Fermi Basis Reference(s):

N/A

Barrier: Reactor Coolant System

Category: 4. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

A. CHRRM reading > 8.72E+1 R/hr

Definition(s):

None

Basis:

Plant-Specific

For Fermi 2, the Containment High Range Radiation Monitor (CHRRM) is used to measure drywell radiation levels. A valid CHRRM reading of 8.72E+1 R/hr corresponds to normal coolant activity discharged instantaneously into containment atmosphere (Ref. 1).

Generic

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 4.A since it indicates a loss of the RCS Barrier only.

There is no Potential Loss threshold associated with Primary Containment Radiation.

Fermi Basis Reference(s):

1. EP-EALCALC-FERMI-1402 Fermi EAL Technical Bases Calculations – CHRRM Series Rev. 0
2. NEI 99-01 Primary Containment Radiation RCS Loss 4.A

Barrier: Reactor Coolant System

Category: 4. PC Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Definition(s):

None

Basis:

Plant-Specific

N/A

Generic

N/A

Fermi Basis Reference(s):

N/A

Barrier: Reactor Coolant System

Category: 5. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Definition(s):

None

Basis:

Plant-Specific

N/A

Generic

N/A

Fermi Basis Reference(s):

N/A

Barrier: Reactor Coolant System

Category: 5. PC Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Definition(s):

None

Basis:

Plant-Specific

N/A

Generic

N/A

Fermi Basis Reference(s):

N/A

Barrier: Reactor Coolant System

Category: 6. ED Judgment

Degradation Threat: Loss

Threshold:

A. **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier

Definition(s):

None

Basis:

Plant-Specific

None

Generic

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

Fermi Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Loss 6.A

Barrier: Reactor Coolant System

Category: 6. ED Judgment

Degradation Threat: Potential Loss

Threshold:

A. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier

Definition(s):

None

Basis:

Plant-Specific

None

Generic

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

Fermi Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Potential Loss 6.A

Barrier: Primary Containment

Category: 1. RPV Water Level

Degradation Threat: Loss

Threshold:

None

Definition(s):

None

Basis:

Plant-Specific

N/A

Generic

N/A

Fermi Basis Reference(s):

N/A

Barrier: Primary Containment

Category: 1. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

A. Inadequate core cooling as indicated by ANY of the following:

1. RPV water level **cannot** be restored and maintained > -48 in. with ≥ 5725 gpm Core Spray loop flow

   OR

2. RPV water level **cannot** be restored and maintained > -25 in. with < 5725 gpm Core Spray loop flow

   OR

3. RPV water level **cannot** be determined and core damage is occurring

Definition(s):

None

Basis:

Plant-Specific

Requirements for entry into the Severe Accident Guidelines (SAGs) are established in EOP RPV Control, EOP RPV Control - ATWS and EOP RPV Flooding & EOP RPV Flooding - ATWS (Ref. 1, 2, 3, 4, 5). 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.

Direction is specified that SAG entry is required when:

- EOP RPV Control - RPV water level cannot be restored and maintained above -48 in. with required Core Spray flow.
- EOP RPV Control - RPV water level cannot be restored and maintained above -25 in. (MSCRWL) with insufficient Core Spray flow.
- EOP RPV Control - ATWS - RPV water level cannot be restored and maintained above -25 in. (MSCRWL)

- EOP Flooding & EOP Flooding - ATWS - RPV water level cannot be determined and it is determined that core damage is occurring.

This threshold is also a Loss of the Fuel Clad barrier (FC Loss 1.A). Since SAG entry occurs after core uncovery has occurred, a Loss of the RCS barrier exists (RCS Loss 1.A). SAG entry, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

Generic

The Potential Loss threshold is identical to the Fuel Clad Loss RPV Water Level threshold 2.A. The Potential Loss requirement of SAG entry indicates adequate core cooling cannot be restored and maintained and that core damage is possible. 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.

Fermi Basis Reference(s):

1. EOP 29.100.01 Sheet 1 RPV Control
2. EOP 29.100.01A Sheet 1 RPV Control - ATWS
3. EOP 29.100.01 Sheet 3 RPV Flooding, Emerg. Depress. & Steam Cooling
4. EOP 29.100.01 Sheet 3A RPV Flooding, Emerg. Depress. & Steam Cooling
5. EOP Support Documentation Section 1 Plant Specific Technical Guideline
6. NEI 99-01 RPV Water Level PC Potential Loss 2.A

Barrier: Primary Containment

Category: 2. RCS Leak Rate

Degradation Threat: Loss

Threshold:

A. UNISOLABLE primary system leakage into Secondary Containment that results in exceeding ANY of the following:

1. One or more Secondary Containment Control **Max Safe** Operating Temperatures (EOP Table 12)

   OR

2. Exceeding Secondary Containment Control **Max Safe** Operating Area Radiation Level on channel 14 ARM (RBSB Torus Room)

Definition(s):

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

Basis:

Plant-Specific

The presence of elevated general area temperatures or radiation levels in the Reactor Building (RB) may be indicative of UNISOLABLE primary system leakage outside the Primary Containment. When parameters reach the threshold 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. Determination of whether the leak is isolated to preclude EAL declaration must occur within the 15-minute assessment period. (Ref. 1, 2)

The use of secondary containment radiation monitors provides indication of increased release that may be indicative of a challenge to Primary Containment. The Secondary Containment Control EOP **Max Safe** area radiation values have been selected because these values are easily recognizable and have a defined basis.

The only Secondary Containment Maximum Safe Operating Radiation Level that can be directly obtained in the Control Room is the RBSB Torus Room on ARM Channel 14. Other installed area radiation monitors listed in the EOP **Max Safe** table are NOT used in this determination because of instrument range limitations that would require a survey to determine the EOP referenced **Max Safe** value. Surveys initiated for EOP **Max Safe** assessments do not generally provide timely information to determine classification of this event. The other areas listed as part of the EOP **Max Safe** radiation areas that would require a radiation survey to verify the **Max Safe** radiation values are bound in classification consideration by the installed EOP **Max Safe** temperature instrumentation, and Primary Containment LOSS 5.A (UNISOLABLE direct downstream pathway to the environment exists after primary containment isolation signal).

In general, multiple indications should be used to determine if a primary system is discharging outside Primary Containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the secondary containment since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room flooding, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the secondary containment.

<u>Generic</u>

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 2.A this threshold would result in a Site Area Emergency.

There is no Potential Loss threshold associated with Primary Containment Isolation Failure.

Fermi Basis Reference(s):

1. EOP Support Documentation Section 1 Plant Specific Technical Guideline
2. EOP 29.100.01 Sheet 5 Secondary Containment and Rad Release
3. NEI 99-01 RCS Leak Rate PC Loss 3.C

Barrier: Primary Containment

Category: 2. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Definition(s):

None

Basis:

Plant-Specific

N/A

Generic

N/A

Fermi Basis Reference(s):

N/A

Barrier: Primary Containment

Category: 3. PC Conditions

Degradation Threat: Loss

Threshold:

A. 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.

Basis:

Plant-Specific

None

Generic

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.

Fermi Basis Reference(s):

1. NEI 99-01 Primary Containment Conditions PC Loss 1.A

Barrier: Primary Containment

Category: 3. PC Conditions

Degradation Threat: Loss

Threshold:

B. Primary Containment pressure response not consistent with LOCA conditions

Definition(s):

None

Basis:

Plant-Specific

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.

USAR Sections 6.2.1.3.2 and 6.2.1.3.3 provide a summary of Primary Containment pressure response for the design basis loss of coolant accident and the conditions resulting in the release of RCS inventory to the containment (Ref. 1, 3). The maximum calculated drywell pressure is approximately 50 psig and then stabilizes at approximately 30 psig with torus pressure at approximately 25 psig 30 seconds after the break (Ref. 2). These pressures are well below the design allowable drywell pressure of 62 psig. (Ref. 1)

Due to conservatisms in LOCA analyses, actual pressure response is expected to be less than the analyzed response. LOCA conditions are manifested on Control Room instrumentation by drywell pressure rising with torus pressure following and eventually equalizing (around 18 psig for the DBA LOCA) (Ref. 3, 4).

Generic

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.

Fermi Basis Reference(s):

1. USAR Section 6.2.1.3.2 Recirculation Line Break - Short Term Response
2. USAR Figure 6.2-9 Recirculation Line Break Primary Containment Initial Pressure Transient (3499 MWT)
3. USAR Section 6.2.1.3.3 Recirculation Line Break - Long Term Response
4. USAR Figure 6.2-11 Primary Containment Pressure Long Term Response (3499 MWT)
5. NEI 99-01 Primary Containment Conditions PC Loss 1.B

Barrier: Primary Containment

Category: 3. PC Conditions

Degradation Threat: Potential Loss

Threshold:

A. Primary Containment Pressure > 62 psig

Definition(s):

None

Basis:

Plant-Specific

The Primary Containment pressure of 62 psig is based on the primary containment design pressure (Ref. 1).

Generic

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.

Fermi Basis Reference(s):

1. UFSAR Table 6.2-4 Drywell to Suppression Chamber Vacuum Breaker Valve Data
2. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.A

Barrier: Primary Containment

Category: 3. PC Conditions

Degradation Threat: Potential Loss

Threshold:

B. > 6% H_2 AND > 5% O_2 in EITHER the drywell or suppression chamber

Definition(s):

None

Basis:

Plant-Specific

Explosive (deflagration) mixtures in the Primary Containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/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, Ref. 1) and readily recognizable because 6% hydrogen is well above the EOP Primary Containment Control entry condition of 1% (Ref. 1, 2).

Generic

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.

Fermi Basis Reference(s):

1. EOP Support Documentation Section 1 Plant Specific Technical Guideline
2. EOP 29.100.01 Sheet 2 Primary Containment Control
3. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.B

Barrier: Primary Containment

Category: 3. PC Conditions

Degradation Threat: Potential Loss

Threshold:

C. EOP Heat Capacity Limit (HCL) exceeded

Definition(s):

None

Basis:

Plant-Specific

The Heat Capacity Limit (HCL) is given in EOP Curves, Cautions and Tables (Ref. 1).

Generic

The Heat Capacity Temperature Limit (HCL) is the highest torus water 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 HCL is a function of RPV pressure, torus water temperature and torus 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.

Fermi Basis Reference(s):

1. EOP 29.100.01 Sheet 6 Curves, Cautions and Tables

2. EOP 29.100.01 Sheet 2 Primary Containment Control
3. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.C

Barrier: Primary Containment

Category: 4. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

None

Definition(s):

None

Basis:

Plant-Specific

N/A

Generic

N/A

Fermi Basis Reference(s):

N/A

Barrier: Primary Containment

Category: 4. PC Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

A. CHRRM reading > 1.79E+4 R/hr

Definition(s):

None

Basis:

Plant-Specific

The "potential loss" EAL value corresponds to at least 20% clad damage with release into the primary containment. This EAL corresponds to loss of both the Fuel Clad and RCS barriers with Potential Loss of the Primary Containment barrier, and would result in declaration of a General Emergency.

For Fermi 2, the Containment High Range Radiation Monitor (CHRRM) is used to measure drywell radiation levels. A valid CHRRM reading of 1.79E+4 R/hr corresponds to 20% gap release discharged instantaneously into containment atmosphere (Ref. 1).

Generic

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.

Fermi Basis Reference(s):

1. EP-EALCALC-FERMI-1402 Fermi EAL Technical Bases Calculations – CHRRM Series Rev. 0
2. NEI 99-01 Primary Containment Radiation PC Potential Loss 4.A

Barrier: Primary Containment

Category: 5. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

A. 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.

Basis:

Plant-Specific

This Primary Containment isolation failure threshold is based on failure to successfully isolate the Primary Containment following a valid isolation signal (RPV water level 1, 2 or 3 or high drywell pressure) resulting in a direct downstream pathway to the environment for any of the following containment isolation signals (Ref. 1):

- Group 1 - Main Steam System
- Group 2 - Reactor Water Sample System
- Group 4 - RHR Shutdown Cooling and Head Vent
- Group 6 – HPCI Steam Supply Line
- Group 8 – RCIC Steam Supply Line
- Group 10/11 - Reactor Water Cleanup System Inboard/Outboard
- Group 13 - Drywell Sumps
- Group 14 - Drywell and Suppression Pool Ventilation

These systems, protected by the Primary Containment Isolation System, provide potential direct (non-liquid interfacing system) release pathways from the RCS or Primary Containment atmosphere to the environment should the isolation be unsuccessful.

Determination of whether the leak is isolated to preclude EAL declaration must occur within the 15-minute assessment period. (Ref. 1)

Generic

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 RICs.

Fermi Basis Reference(s):

1. AOP 20.000.21 Reactor Scram Attachment 1 Isolations and Actuations
2. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.A

Barrier: Primary Containment

Category: 5. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

B. Intentional Primary Containment venting, irrespective of offsite radioactivity release rates, per EOPs

Definition(s):

None

Basis:

Plant-Specific

EOP 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, 2). The threshold is met when the operator begins venting the Primary Containment in accordance with 29.ESP.07 , not when actions are taken to bypass interlocks prior to opening the vent valves (Ref. 3). Purge and vent actions specified in EOP Primary Containment Control step PCP-1 to control Primary Containment pressure below the drywell high pressure scram setpoint do not meet this threshold because such action is only permitted if offsite radioactivity release rates will remain below the ODCM limits (Ref. 1, 2, 3).

Generic

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.

Fermi Basis Reference(s):

1. EOP Support Documentation Section 1 Plant Specific Technical Guideline
2. EOP 29.100.01 Sheet 2 Primary Containment Control
3. 29.ESP.07 Primary Containment Venting
4. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.B

Barrier: Primary Containment

Category: 6. ED Judgment

Degradation Threat: Loss

Threshold:

A. **Any** condition in the opinion of the Emergency Director that indicates loss of the Primary Containment barrier

Definition(s):

None

Basis:

Plant-Specific

None

Generic

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

Fermi Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

Barrier: Containment

Category: 5. ED Judgment

Degradation Threat: Potential Loss

Threshold:

A. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the Primary Containment barrier

Definition(s):

None

Basis:

Plant-Specific

None

Generic

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

Fermi Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Potential Loss 6.A