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August 30, 2021

50.54(q)
72.44(f)

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

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

Dear Sir or Madam:

The purpose of this letter is to report a change to Emergency Plan Implementing Procedure (EPIP) 5.7.1, Emergency Classification (Revision 70), and provide a summary of the associated 10 CFR 50.54(q) analysis for the changes to the EPIP.

This letter contains no regulatory commitments. If you have any questions regarding this submittal, please contact me at (402) 825-5416.

Sincerely,

Linda Dewhirst
Regulatory Affairs and Compliance Manager

/bk

Attachment: Report of Change and Summary of 50.54(q) Analysis
Emergency Plan Implementing Procedure 5.7.1, Revision 70

Enclosure: Emergency Plan Implementing Procedure 5.7.1, Revision 70

cc: Regional Administrator, w/attachment and enclosure
USNRC – Region IV

Director, Division of Fuel Management, w/attachment and enclosure
Office of Nuclear Material Safety and Safeguards

Senior Resident Inspector, w/attachment and enclosure
USNRC – CNS

NPG Distribution, w/attachment and w/o enclosure

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ATTACHMENT

Cooper Nuclear Station
Report of Change and Summary of 50.54(q) Analysis
Emergency Plan Implementing Procedure 5.7.1, Revision 70

EPIP 5.7.1, Emergency Classification, Revision 70

Change Description:

Emergency Plan Implementing Procedure (EPIP) 5.7.1, Emergency Classification, was revised as follows:

- Revised the Emergency Action Level (EAL) Classification Matrix Wallchart to allow for Table C-7 to be visible on the Cold SD/Refueling System Malfunction portion of the matrix. This table was inadvertently hidden behind another table in a previous revision.
- Expanded the Notes Table in Attachment 4, EAL Classification Matrix Hardcard for "Hot" 'S' System Malfunction to allow Notes 1 and 8 to be fully legible.
- Updated EAL Classification Matrix revision number in Attachment 4.
- Changed value at which instrumentation actuates for recording a seismic event from 0.1 g to 0.01 g in Attachment 2, EAL Technical Bases, that was inappropriately revised in the last revision of the EPIP.

The changes made to the EPIP 5.7.1 described above are considered editorial and/or formatting changes. As such, completion of a full 10 CFR 50.54(q) evaluation was not required.

ENCLOSURE

Cooper Nuclear Station
Emergency Plan Implementing Procedure 5.7.1, Revision 70

COOPER NUCLEAR STATION

Operations Manual

Emergency Preparedness

EMERGENCY PLAN IMPLEMENTING PROCEDURE

5.7.1

EMERGENCY CLASSIFICATION

Level of Use:	**MULTIPLE**
Quality:	**QAPD RELATED**
Effective Date:	**8/4/21**
Approval Authority:	**ITR-RDM**
Procedure Owner:	**EMERG PREP DRILL SCENARIO COORD**

TABLE OF CONTENTS

1. ENTRY CONDITIONS [REFERENCE USE]

   1.1 An Emergency Operation Procedure has been initiated; or

   1.2 An unusual occurrence has taken place at or near site.

2. INSTRUCTIONS [REFERENCE USE]

   2.1 **PERFORM** classification and declaration activities per EPIP 5.7.1.1 as supported by attachments in this procedure.

ATTACHMENT 1 EAL SCHEME EXPLANATION AND RATIONALE [INFORMATION USE]

1. PURPOSE

1.1 This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Cooper Nuclear Station (CNS). It should be used to facilitate review of the CNS EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of Emergency Plan Implementing Procedure 5.7.1, Emergency Classification, 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.

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

1.3 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 10CFR50.54(q).

2. DISCUSSION

2.1 BACKGROUND

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

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

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

2.1.3.1 Consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur during plant shutdown conditions.

2.1.3.2 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).

2.1.3.3 Simplifying the fission product barrier EAL threshold for a Site Area Emergency.

2.1.4 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 (ref. 3.1.1), CNS conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 FISSION PRODUCT BARRIERS

2.2.1 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.

2.2.2 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 a greater probability of barrier loss and reduced certainty of maintaining the barrier.

2.2.3 The primary fission product barriers are:

2.2.3.1 Fuel Clad (FC): The Fuel Clad barrier consists of the cladding material that contains the fuel pellets.

2.2.3.2 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.

2.2.3.3 Primary Containment (PC): The Primary Containment barrier includes the drywell, the wetwell (torus), their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. 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 EMERGENCY CLASSIFICATION BASED ON FISSION PRODUCT BARRIER DEGRADATION

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

2.3.1.1 Alert: Any loss or any potential loss of either the Fuel Clad or RCS Barrier

2.3.1.2 Site Area Emergency: Loss or potential loss of any two barriers

2.3.1.3 General Emergency: Loss of any two barriers and loss or potential loss of the third barrier.

2.4 EAL RELATIONSHIP TO EOPS

2.4.1 Where possible, the EALs have been made consistent with and utilize the conditions defined in the CNS Emergency Operating Procedures (EOPs). While the symptoms that drive Operator actions specified in the EOPs are not indicative of all possible conditions which warrant emergency classification, they define the symptoms, independent of initiating events, for which reactor plant safety and/or fission product barrier integrity are threatened. When these symptoms are clearly representative of one of the NEI Initiating Conditions, they have been utilized as an EAL. This permits rapid classification of emergency situations based on plant conditions without the need for additional evaluation or event diagnosis. Although some of the EALs presented here are based on conditions defined in the EOPs, classification of emergencies using these EALs is not dependent upon EOP entry or execution. The EALs can be utilized independently or in conjunction with the EOPs.

2.5 SYMPTOM-BASED VS. EVENT BASED APPROACH

2.5.1 To the extent possible, the EALs are symptom based. That is, the action level threshold is defined by values of key plant operating parameters that identify emergency or potential emergency conditions. This approach is appropriate because it allows the full scope of variations in the types of events to be classified as emergencies. However, a purely symptom-based approach is not sufficient to address all events for which emergency classification is appropriate. Particular events to which no pre-determined symptoms can be ascribed have also been utilized as EALs since they may be indicative of potentially more serious conditions not yet fully realized.

2.6 EAL ORGANIZATION

2.6.1 The CNS EAL scheme includes the following features:

2.6.1.1 Division of the EAL set into three broad groups:

a. GROUPS

1. EALs applicable under any plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.

2. 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.

3. 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.

b. 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.

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

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

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

EAL GROUPS, CATEGORIES AND SUBCATEGORIES

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

2.7 TECHNICAL BASES INFORMATION

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

2.7.1.1 CATEGORY LETTER AND TITLE:

2.7.1.2 SUBCATEGORY NUMBER AND TITLE:

2.7.1.3 INITIATING CONDITION (IC):

a. Site-specific description of the generic IC given in NEI 99-01, Revision 6.

2.7.1.4 EAL IDENTIFIER (ENCLOSED IN RECTANGLE):

a. 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 (A, C, E, F, H or S)

2. Second character (letter): The emergency classification (G, S, A or U)

a) G = General Emergency
S = Site Area Emergency
A = Alert
U = Unusual Event

3. Third character (number): Subcategory number within the given category.

a) Subcategories are sequentially numbered beginning with the number one (1). If a category does not have a subcategory, this character is assigned the number one (1).

4. Fourth character (number): The numerical sequence of the EAL within the EAL subcategory.

   a) If the subcategory has only one EAL, it is given the number one (1).

2.7.1.5 CLASSIFICATION (ENCLOSED IN RECTANGLE):

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

2.7.1.6 EAL (ENCLOSED IN RECTANGLE):

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

2.7.1.7 MODE APPLICABILITY:

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

2.7.1.8 DEFINITIONS:

a. 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.

2.7.1.9 BASIS:

a. An EAL basis section that provides CNS-relevant information concerning the EAL as well as a description of the rationale for the EAL as provided in NEI 99-01, Revision 6.

2.7.1.10 REFERENCE(S):

a. Source documentation from which the EAL is derived

2.8 OPERATING MODE APPLICABILITY

2.8.1 MODES

2.8.1.1 POWER OPERATION (MODE 1)

a. Reactor mode switch is in RUN.

2.8.1.2 STARTUP (MODE 2)

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

2.8.1.3 HOT SHUTDOWN (MODE 3)

a. The reactor mode switch is in SHUTDOWN and average reactor coolant temperature is > 212°F.

2.8.1.4 COLD SHUTDOWN (MODE 4)

a. The reactor mode switch is in SHUTDOWN and average reactor coolant temperature is ≤ 212°F.

2.8.1.5 REFUELING (MODE 5)

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

2.8.1.6 DEFUELED (MODE DEF)

a. RPV contains no irradiated fuel.

2.8.2 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.

2.9 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

2.9.1 GENERAL CONSIDERATIONS

2.9.1.1 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 Category F matrices, EALs are based on loss or potential loss of Fission Product Barrier Thresholds.

2.9.1.2 EAL matrices should be read from left to right, from General Emergency to Unusual Event, and top to bottom. Declaration decisions should be independently verified before declaration is made except when gaining this verification would exceed the 15-minute declaration requirement. Place keeping should be used on all EAL matrices.

2.10 CLASSIFICATION TIMELINESS

2.10.1 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. 3.1.8).

2.11 VALID INDICATIONS

2.11.1 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.

2.11.2 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.

2.12 IMMINENT CONDITIONS

2.12.1 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.

2.13 PLANNED VS. UNPLANNED EVENTS

2.13.1 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 10CFR50.72 (ref. 3.1.4).

2.14 CLASSIFICATION BASED ON ANALYSIS

2.14.1 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).

2.15 EMERGENCY DIRECTOR JUDGMENT

2.15.1 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

2.16 CLASSIFICATION METHODOLOGY

2.16.1.1 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 15 minutes after the process "clock" started.

2.16.1.2 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. 3.1.8).

2.17 CLASSIFICATION OF MULTIPLE EVENTS AND CONDITIONS

2.17.1 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:

2.17.2 If an Alert EAL and a Site Area Emergency EAL are met, whether at one unit or at two units, a Site Area Emergency should be declared.

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

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

2.18 CONSIDERATION OF MODE CHANGES DURING CLASSIFICATION

2.18.1 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.

2.18.2 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.

2.19 CLASSIFICATION OF IMMINENT CONDITIONS

2.19.1 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.

2.20 EMERGENCY CLASSIFICATION LEVEL UPGRADING AND TERMINATION

2.20.1 An ECL may be terminated when the event or condition that meets the classified IC and EAL no longer exists, and other site-specific termination requirements are met.

2.21 CLASSIFICATION OF SHORT-LIVED EVENTS

2.21.1 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.

2.22 CLASSIFICATION OF TRANSIENT CONDITIONS

2.22.1 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.

2.22.2 EAL momentarily met during expected plant response – In instances in which 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.

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

2.22.4 An ATWS occurs and the high pressure ECCS systems fail to automatically start. 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.

2.22.5 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.

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

2.23.1 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.

2.23.1.1 In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 (ref. 3.1.3) is applicable. Specifically, the event should be reported to the NRC per 10CFR50.72 (ref. 3.1.4) 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.

2.24 RETRACTION OF AN EMERGENCY DECLARATION

2.24.1 Guidance on the retraction of an emergency declaration reported to the NRC is discussed in NUREG-1022 (ref. 3.1.3).

2.25 CNS-TO-NEI 99-01, Revision 6 EAL CROSS-REFERENCE

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

CNS	**NEI 99-01, Revision 6**	
EAL	**IC**	**Example EAL**
AU1.1	AU1	1, 2
AU1.2	AU1	3
AU2.1	AU2	1
AA1.1	AA1	1

CNS	NEI 99-01, Revision 6	
EAL	**IC**	**Example EAL**
AA1.2	AA1	2
AA1.3	AA1	3
AA1.4	AA1	4
AA2.1	AA2	1
AA2.2	AA2	2
AA2.3	AA2	3
AA3.1	AA3	1
AA3.2	AA3	2
AS1.1	AS1	1
AS1.2	AS1	2
AS1.3	AS1	3
AS2.1	AS2	1
AG1.1	AG1	1
AG1.2	AG1	2
AG1.3	AG1	3
AG2.1	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, 2
CA6.1	CA6	1

CNS	NEI 99-01, Revision 6	
EAL	**IC**	**Example EAL**
CS1.1	CS1	1
CS1.2	CS1	2
CS1.3	CS1	3
CG1.1	CG1	1
CG1.2	CG1	2
EU1.1	E-HU1	1
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1, 2, 3
HU2.1	HU2	1
HU3.1	HU3	1
HU3.2	HU3	2
HU3.3	HU3	3
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
N/A	HG1	1

CNS	NEI 99-01, Revision 6	
EAL	**IC**	**Example EAL**
HG7.1	HG7	1
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	1
SU4.2	SU3	2
SU5.1	SU4	1, 2, 3
SU6.1	SU5	1
SU6.2	SU5	2
SU7.1	SU6	1, 2, 3
N/A	SU7	1, 2
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

ATTACHMENT 2 EMERGENCY ACTION LEVEL TECHNICAL BASES [INFORMATION USE]

Category A – Abnormal Rad Levels/Rad Effluent

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

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

At lower levels, abnormal radioactivity releases may be indicative of a failure of containment systems or precursors to more significant releases. At higher release rates, offsite radiological conditions may result which require offsite protective actions. Elevated area radiation levels in 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: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AU1.1 Unusual Event

Reading on **any** Table A-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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is 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 due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

(continued on next page)

Table A-1 Effluent Monitor Classification Thresholds

	Release Point	GE ≥ 15 min.	SAE ≥ 15 min.	Alert ≥ 15 min.	UE ≥ 60 min.
Gaseous	**ERP** RMP-RM-3A/B	2.58E+9 µCi/sec [N/A]*	2.58E+8 µCi/sec [N/A]*	2.58E+7 µCi/sec [1.76E+7 µCi/sec]*	N/A [2.24E+6 µCi/sec]*
Gaseous	**Rx Bldg Vent** RMV-RM-40	8.86E+6 µCi/sec [N/A]*	8.86E+5 µCi/sec [N/A]*	8.85E+4 µCi/sec [1.77E+6 µCi/sec]*	N/A [8.48E+4 µCi/sec]*
Gaseous	**Turb Bldg Vent** RMV-RM-20A/B	4.43E+6 µCi/sec [N/A]*	4.43E+5 µCi/sec [N/A]*	4.43E+4 µCi/sec [1.77E+6 µCi/sec]*	N/A [9.02E+4 µCi/sec]*
Gaseous	**RW / ARW Bldg Vent** RMP-RM-30A/B	4.43E+6 µCi/sec [N/A]*	4.43E+5 µCi/sec [N/A]*	4.43E+4 µCi/sec [1.77E+6 µCi/sec]*	N/A [9.08E+4 µCi/sec]*
Liquid	**Rad Waste Effluent** RMP-RM-354	N/A	N/A	N/A	2 X alarm setpoint
Liquid	**Service Water Effluent** RMP-RM-351A/B	N/A	N/A	N/A	1.25E-4 µCi/cc

* Bracketed values are **only** applicable for a non-degraded core per EPIP 5.7.17 Attachment 5

Mode Applicability:

All

Definition(s):

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.

Basis:

This EAL 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.

(continued on next page)

The Table A-1 column "UE" gaseous [bracketed] effluent thresholds are **only** applicable when it has been determined that a degraded core condition does **not** exist (ref. 2, 3).

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 as well as radioactivity releases that cause effluent radiation monitor readings to exceed 2 times the limit established by a radioactivity discharge permit. Such releases are typically associated with planned batch releases from non-continuous release pathways (e.g., radwaste, waste gas).

The Table A-1 effluent threshold for the liquid radwaste effluent monitor RMP-RM-354 has been implemented as "2 x alarm setpoint" rather than the calculated value of 1.36E-2 μCi/sec for operational considerations (ref. 2).

Escalation of the emergency classification level would be via IC AA1.

Reference(s):

1. Off-Site Dose Assessment Manual - ODAM - For Assessment of Gaseous and Liquid Effluents at Cooper Nuclear Station.
2. Calculation No. 19-011 EAL Rev. 6 Threshold Calculations (EP-CALC-CNS-1901 Radiological Effluent EAL Threshold Values).
3. EPIP 5.7.17 CNS-DOSE Assessment.
4. NEI 99-01 AU1.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AU1.2 Unusual Event

Sample analysis for a gaseous or liquid release indicates a concentration or release rate ≥ 2 x ODAM limits for ≥ 60 min. (Notes 1, 2)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is 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:

This EAL 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.

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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 AA1.

Reference(s):

1. Off-Site Dose Assessment Manual - ODAM - For Assessment of Gaseous and Liquid Effluents at Cooper Nuclear Station.
2. NEI 99-01 AU1.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 0.01 Rem TEDE or 0.05 Rem thyroid CDE

EAL:

AA1.1 Alert

Reading on **any** Table A-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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is 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 due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

(continued on next page)

Table A-1 Effluent Monitor Classification Thresholds

	Release Point	GE ≥ 15 min.	SAE ≥ 15 min.	Alert ≥ 15 min.	UE ≥ 60 min.
Gaseous	**ERP** RMP-RM-3A/B	2.58E+9 µCi/sec [N/A]*	2.58E+8 µCi/sec [N/A]*	2.58E+7 µCi/sec [1.76E+7 µCi/sec]*	N/A [2.24E+6 µCi/sec]*
Gaseous	**Rx Bldg Vent** RMV-RM-40	8.86E+6 µCi/sec [N/A]*	8.86E+5 µCi/sec [N/A]*	8.85E+4 µCi/sec [1.77E+6 µCi/sec]*	N/A [8.48E+4 µCi/sec]*
Gaseous	**Turb Bldg Vent** RMV-RM-20A/B	4.43E+6 µCi/sec [N/A]*	4.43E+5 µCi/sec [N/A]*	4.43E+4 µCi/sec [1.77E+6 µCi/sec]*	N/A [9.02E+4 µCi/sec]*
Gaseous	**RW / ARW Bldg Vent** RMP-RM-30A/B	4.43E+6 µCi/sec [N/A]*	4.43E+5 µCi/sec [N/A]*	4.43E+4 µCi/sec [1.77E+6 µCi/sec]*	N/A [9.08E+4 µCi/sec]*
Liquid	**Rad Waste Effluent** RMP-RM-354	N/A	N/A	N/A	2 X alarm setpoint
Liquid	**Service Water Effluent** RMP-RM-351A/B	N/A	N/A	N/A	1.25E-4 µCi/cc

* Bracketed values are **only** applicable for a non-degraded core per EPIP 5.7.17 Attachment 5

Mode Applicability:

All

Definition(s):

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.

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Basis:

This EAL 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).

The Table A-1 column "ALERT" gaseous [bracketed] effluent thresholds are **only** applicable when it has been determined that a degraded core condition does **not** exist (ref. 1, 2).

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 AS1.

Reference(s):

1. Calculation No. 19-011 EAL Rev. 6 Threshold Calculations (EP-CALC-CNS-1901 Radiological Effluent EAL Threshold Values).
2. EPIP 5.7.17 CNS-DOSE Assessment.
3. NEI 99-01 AA1.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 0.01 Rem TEDE or 0.05 Rem thyroid CDE

EAL:

AA1.2 Alert

Dose assessment using actual meteorology indicates doses > 0.01 Rem TEDE or 0.05 Rem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 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 boundary defined by a 1-mile radius around the plant.

Basis:

This EAL 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 AS1.

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Reference(s):

1. EPIP 5.7.17, CNS-DOSE Assessment, and/or EPIP 5.7.17.1, Dose Assessment (Manual).
2. NEI 99-01 AA1.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 0.01 Rem TEDE or 0.05 Rem thyroid CDE

EAL:

AA1.3 Alert

Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 0.01 Rem TEDE or 0.05 Rem 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is 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 boundary defined by a 1-mile radius around the plant.

Basis:

This EAL 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.

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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.

This EAL is assessed per the ODAM (ref.1)

Escalation of the emergency classification level would be via IC AS1.

Reference(s):

1. EPIP 5.7.17, CNS-DOSE Assessment, and/or EPIP 5.7.17.1, Dose Assessment (Manual).
2. NEI 99-01 AA1.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 0.01 Rem TEDE or 0.05 Rem thyroid CDE

EAL:

AA1.4 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is 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 boundary defined by a 1-mile radius around the plant.

Basis:

This EAL 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).

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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 AS1.

Reference(s):

1. EPIP 5.7.18 Off-Site and Site Boundary Monitoring.
2. NEI 99-01 AA1.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 0.10 Rem TEDE or 0.50 Rem thyroid CDE

EAL:

AS1.1 Site Area Emergency

Reading on **any** Table A-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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is 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 due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

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Table A-1 Effluent Monitor Classification Thresholds

	Release Point	GE ≥ 15 min.	SAE ≥ 15 min.	Alert ≥ 15 min.	UE ≥ 60 min.
Gaseous	**ERP** RMP-RM-3A/B	2.58E+9 µCi/sec [N/A]*	2.58E+8 µCi/sec [N/A]*	2.58E+7 µCi/sec [1.76E+7 µCi/sec]*	N/A [2.24E+6 µCi/sec]*
Gaseous	**Rx Bldg Vent** RMV-RM-40	8.86E+6 µCi/sec [N/A]*	8.86E+5 µCi/sec [N/A]*	8.85E+4 µCi/sec [1.77E+6 µCi/sec]*	N/A [8.48E+4 µCi/sec]*
Gaseous	**Turb Bldg Vent** RMV-RM-20A/B	4.43E+6 µCi/sec [N/A]*	4.43E+5 µCi/sec [N/A]*	4.43E+4 µCi/sec [1.77E+6 µCi/sec]*	N/A [9.02E+4 µCi/sec]*
Gaseous	**RW / ARW Bldg Vent** RMP-RM-30A/B	4.43E+6 µCi/sec [N/A]*	4.43E+5 µCi/sec [N/A]*	4.43E+4 µCi/sec [1.77E+6 µCi/sec]*	N/A [9.08E+4 µCi/sec]*
Liquid	**Rad Waste Effluent** RMP-RM-354	N/A	N/A	N/A	2 X alarm setpoint
Liquid	**Service Water Effluent** RMP-RM-351A/B	N/A	N/A	N/A	1.25E-4 µCi/cc

* Bracketed values are **only** applicable for a non-degraded core per EPIP 5.7.17, Attachment 5

Mode Applicability:

All

Definition(s):

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.

(continued on next page)

Basis:

This EAL 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 AG1.

Reference(s):

1. Calculation No. 19-011 EAL Rev. 6 Threshold Calculations (EP-CALC-CNS-1901 Radiological Effluent EAL Threshold Values).
2. NEI 99-01 AS1.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 0.10 Rem TEDE or 0.50 Rem thyroid CDE

EAL:

AS1.2 Site Area Emergency

Dose assessment using actual meteorology indicates doses > 0.10 Rem TEDE or 0.50 Rem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 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 boundary defined by a 1-mile radius around the plant.

Basis:

This EAL 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 AG1.

Reference(s):

1. EPIP 5.7.17, CNS-DOSE Assessment, and/or EPIP 5.7.17.1, Dose Assessment (Manual).
2. NEI 99-01 AS1.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 0.10 Rem TEDE or 0.50 Rem thyroid CDE

EAL:

AS1.3 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is 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 boundary defined by a 1-mile radius around the plant.

Basis:

This EAL 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.

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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 AG1.

Reference(s):

1. EPIP 5.7.18 Off-Site and Site Boundary Monitoring.
2. NEI 99-01 AS1.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1.00 Rem TEDE or 5.00 Rem thyroid CDE

EAL:

AG1.1 General Emergency

Reading on **any** Table A-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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is 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 due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

(continued on next page)

Table A-1 Effluent Monitor Classification Thresholds

	Release Point	GE ≥ 15 min.	SAE ≥ 15 min.	Alert ≥ 15 min.	UE ≥ 60 min.
Gaseous	**ERP** RMP-RM-3A/B	2.58E+9 µCi/sec [N/A]*	2.58E+8 µCi/sec [N/A]*	2.58E+7 µCi/sec [1.76E+7 µCi/sec]*	N/A [2.24E+6 µCi/sec]*
Gaseous	**Rx Bldg Vent** RMV-RM-40	8.86E+6 µCi/sec [N/A]*	8.86E+5 µCi/sec [N/A]*	8.85E+4 µCi/sec [1.77E+6 µCi/sec]*	N/A [8.48E+4 µCi/sec]*
Gaseous	**Turb Bldg Vent** RMV-RM-20A/B	4.43E+6 µCi/sec [N/A]*	4.43E+5 µCi/sec [N/A]*	4.43E+4 µCi/sec [1.77E+6 µCi/sec]*	N/A [9.02E+4 µCi/sec]*
Gaseous	**RW / ARW Bldg Vent** RMP-RM-30A/B	4.43E+6 µCi/sec [N/A]*	4.43E+5 µCi/sec [N/A]*	4.43E+4 µCi/sec [1.77E+6 µCi/sec]*	N/A [9.08E+4 µCi/sec]*
Liquid	**Rad Waste Effluent** RMP-RM-354	N/A	N/A	N/A	2 X alarm setpoint
Liquid	**Service Water Effluent** RMP-RM-351A/B	N/A	N/A	N/A	1.25E-4 µCi/cc

* Bracketed values are **only** applicable for a non-degraded core per EPIP 5.7.17 Attachment 5

Mode Applicability:

All

Definition(s):

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.

(continued on next page)

Basis:

This EAL 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.

Reference(s):

1. Calculation No. 19-011 EAL Rev. 6 Threshold Calculations (EP-CALC-CNS-1901 Radiological Effluent EAL Threshold Values).
2. NEI 99-01 AG1.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1.00 Rem TEDE or 5.00 Rem thyroid CDE

EAL:

AG1.2 General Emergency

Dose assessment using actual meteorology indicates doses > 1.00 Rem TEDE or 5.00 Rem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 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 boundary defined by a 1-mile radius around the plant.

Basis:

This EAL 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.

Reference(s):

1. EPIP 5.7.17, CNS-DOSE Assessment, and/or EPIP 5.7.17.1, Dose Assessment (Manual).
2. NEI 99-01 AG1.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1.00 Rem TEDE or 5.00 Rem thyroid CDE

EAL:

AG1.3 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is 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 boundary defined by a 1-mile radius around the plant.

Basis:

This EAL 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.

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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.

Reference(s):

1. EPIP 5.7.18 Off-Site and Site Boundary Monitoring.
2. NEI 99-01 AG1.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: UNPLANNED loss of water level above irradiated fuel

EAL:

AU2.1 Unusual Event

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

- NBI-LI-86 (when calibrated to 1001' elev.)
- FPC-LIT-1, FPC-LIT-2 or FPC-LI-2
- Spent fuel pool low level alarm
- Visual observation

AND

UNPLANNED rise in corresponding area radiation on RMA-RA-1 or RMA-RA-2

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 - Reactor cavity, fuel transfer canal (cattle chute), and spent fuel pool, but **not** including the reactor vessel, comprise the refueling pathway.

Basis:

This EAL 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.

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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 Category C during the Cold Shutdown and Refueling modes.

Normal SFP water level is 37' 6 1/2" above the bottom. A low SFP level alarm can be determined by Annunciator 9-4-2/A-3, FUEL POOL COOLING TROUBLE, alarming due to Annunciator Panel 25-15, Fuel Pool Low Level at 4" below normal. Decreases in SFP water level can also be detected through visual observation. The Skimmer Surge Tank low level alarm (Annunciator 9-4-2/C-3 at 100 ft^3 in the skimmer surge tank, elevation 981' 3") alone may not be conclusive evidence of an uncontrolled loss of inventory from the SFP. SFP weir wall design should prevent inadvertent draining of the SFP through Fuel Pool Cooling and Demineralizer System connections. A Skimmer Surge Tank low level alarm needs to be confirmed by visual observation to determine the extent of inventory loss from the SFP (ref. 1, 3, 4).

During refueling when the RPV head is removed, Shutdown Range RPV water level instrument NBI-LI-86 is recalibrated to read vessel cavity level up to the 1001' elevation (Refuel Floor). With the reactor cavity in communication with the Spent Fuel Pool via the fuel transfer canal, uncontrolled inventory loss can be remotely monitored via this indicator. NBI-LI-86 can be used only if it has been set up to read to 1001' elevation as specified in Procedure 4.6.1, Reactor Vessel Water Level Indication (ref. 5).

Spent Fuel Pool level instruments FPC-LIT-1, FPC-LIT-2 or FPC-LI-2 provide continuous level indication of SFP level from normal level to the top of the spent fuel racks (ref. 6).

Area radiation monitors that may indicate a loss of shielding of spent fuel in the SFP or refueling cavity include (Reference 2, 7):

- RMA-RA-1 (1448) RX BLDG FUEL POOL (HR) AREA.
- RMA-RA-2 (1449) RX BLDG FUEL POOL (LR) AREA.

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Portable radiation monitors are routinely employed to conduct radiation surveys in the Reactor Building. This source of information should not be excluded when considering emergency classification under this EAL, particularly when RMA-RA-1 and RMA-RA-2 may be taken out of service for preventative or corrective maintenance.

Escalation of the emergency classification level would be via IC AA2.

Reference(s):

1. System Operating Procedure 2.2.32, Fuel Pool Cooling and Demineralizer System.
2. Alarm Procedure 2.3_9-3-1, Panel 9-3 - Annunciator 9-3-1.
3. Alarm Procedure 2.3_9-4-2, Panel 9-4 - Annunciator 9-4-2, C-3.
4. Abnormal Procedure 2.4FPC, Fuel Pool Cooling Trouble.
5. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
6. CADD File C0101566, Typical Mounting Arrangement for Spent Fuel Pool Level Instrumentation Sensors at Cooper Nuclear Station (Elevation View).
7. Emergency Procedure 5.1RAD, Building Radiation Trouble.
8. NEI 99-01 AU2.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AA2.1 Alert

IMMINENT uncovery of irradiated fuel in the REFUELING PATHWAY

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY - The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the CNS ISFSI, Confinement Boundary is defined as the NUHOMS Dry Shielded Canister (DSC).

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.

REFUELING PATHWAY - Reactor cavity, fuel transfer canal (cattle chute), and spent fuel pool, but **not** including the reactor vessel, comprise the refueling pathway.

Basis:

This EAL addresses events that have caused IMMINENT or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the REFUELING PATHWAY. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

This EAL escalates from AU2.1 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.

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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 with Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC AS1 .

Reference(s):

1. NEI 99-01 AA2.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity

AND

VALID indication on **EITHER** of the following radiation monitors:

- RMA-RA-1 Fuel Pool Area Rad reading > 5.00E+4 mR/hr
- RMP-RM-452 A-D Rx Bldg Vent Exhaust Plenum Hi-Hi alarm

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY - The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the CNS ISFSI, Confinement Boundary is defined as the NUHOMS Dry Shielded Canister (DSC).

REFUELING PATHWAY - Reactor cavity, fuel transfer canal (cattle chute), and spent fuel pool, but **not** including the reactor vessel, comprise the refueling pathway.

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.

Basis:

This EAL addresses events that have caused actual damage to an irradiated fuel assembly on the refueling floor, Spent Fuel Pool or REFUELING PATHWAY. 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.

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

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).

This EAL is defined by the specific areas where irradiated fuel is located, such as the refueling cavity or Spent Fuel Pool (SFP).

The bases for the ventilation radiation Hi-Hi alarm is a spent fuel handling accident (ref. 2). Fuel Pool area radiation > 5.00E+4 mR/hr represents 100 times the high alarm setpoint (HR) and is unambiguously indicative of spent fuel damage or uncovery (ref. 1).

Escalation of the emergency classification level would be via IC AS1.

Reference(s):

1. Alarm Procedure 2.3_9-3-1, Panel 9-3 - Annunciator 9-3-1, Alarm A-10.
2. Alarm Procedure 2.3_9-4-1, Panel 9-4 - Annunciator 9-4-1, Alarm E-4.
3. NEI 99-01 AA2.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AA2.3 Alert

Lowering of spent fuel pool level to 25.2 ft. (Level 2) on RHR-TR-131, FPC-LIT-1, FPC-LIT-2 or FPC-LI-2

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL addresses events that have caused 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 AS1 or AS2.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2 – 25.2 ft.) and SFP level at the top of the fuel racks (Level 3 – 15.7 ft. [6 in. above the top of the fuel racks]) (ref. 1).

Visual confirmation of SFP level may be used to validate SFP level instrumentation indicators.

Reference(s):

1. CADD File C0101566, Typical Mounting Arrangement for Spent Fuel Pool Level Instrumentation Sensors at Cooper Nuclear Station (Elevation View).
2. NEI 99-01 AA2.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AS2.1 Site Area Emergency

Lowering of spent fuel pool level to 15.7 ft. (Level 3) on RHR-TR-131, FPC-LIT-1, FPC-LIT-2 or FPC-LI-2

Mode Applicability:

All

Definition(s):

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:

This EAL 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 AG1 or AG2.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2 – 25.2 ft.) and SFP level at the top of the fuel racks (Level 3 – 15.7 ft. [6 in. above the top of the fuel racks]) (ref. 1).

Visual confirmation of SFP level may be used to validate SFP level instrumentation indicators.

Reference(s):

1. CADD File C0101566, Typical Mounting Arrangement for Spent Fuel Pool Level Instrumentation Sensors at Cooper Nuclear Station (Elevation View).
2. NEI 99-01 AS2.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AG2.1 General Emergency

Spent fuel pool level **cannot** be restored to at least 15.7 ft. (Level 3) on RHR-TR-131, FPC-LIT-1, FPC-LIT-2 or FPC-LI-2 for ≥ 60 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL 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.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2 – 25.2 ft.) and SFP level at the top of the fuel racks (Level 3 – 15.7 ft. [6 in. above the top of the fuel racks]) (ref. 1).

Visual confirmation of SFP level may be used to validate SFP level instrumentation indicators.

Reference(s):

1. CADD File C0101566, Typical Mounting Arrangement for Spent Fuel Pool Level Instrumentation Sensors at Cooper Nuclear Station (Elevation View).
2. NEI 99-01 AG2.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

AA3.1 Alert

Dose rate > 15 mR/hr in **EITHER** of the following areas:

- Control Room (RMA-RA-20)
- Central Alarm Station (Portable Radiation Monitor local indicator or by survey)

Mode Applicability:

All

Definition(s):

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

Basis:

Areas that meet this threshold include the Control Room (CR) and the Central Alarm Station (CAS). The Control Room is monitored for excessive radiation by one detector, RMA-RA-20. The CAS is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations. There are no permanently installed area radiation monitors in CAS but there is a portable radiation monitor with local indications or local surveys that may be used to assess this EAL threshold. Therefore, this threshold is evaluated using local radiation survey for this area (ref. 1, 2).

This EAL 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 rise in radiation levels and determine if another IC may be applicable.

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

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Reference(s):

1. Alarm Procedure 2.3_9-3-1, Panel 9-3 - Annunciator 9-3-1, B-10.
2. Emergency Procedure 5.1RAD, Building Radiation Trouble.
3. NEI 99-01 AA3.

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

AA3.2 Alert

An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table A-2 room or area (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 A-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Turbine Building 903' Controlled Corridor	1
Reactor Building 958'	3
Reactor Building 931' RHR Hx Room 1B	
Reactor Building 931' RHR Hx Room 1A	
Reactor Building 931' RWCU Hx Room	
Reactor Building 931' General Area	
Reactor Building 903' General Area	
Reactor Building 903' RHR Hx Room 1A	
Reactor Building 903' RHR Hx Room 1B	
Reactor Building 903' Angle Valve Room	
Reactor Building 890' Torus area	
Reactor Building 881' NW Quad	
Control Building 882' ECST room	
Reactor Building 881' Torus Area	4

(continued on next page)

Mode Applicability:

1 – Power Operation, 3 – Hot Shutdown, 4 – Cold Shutdown

Definition(s):

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

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:

This EAL 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 AA3.2, 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 3.
- 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.).

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- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

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

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

The list of plant rooms or areas with entry-related mode applicability identified specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

Reference(s):

1. Attachment 2 Safe Operation & Shutdown Rooms/Areas Tables A-2 & H-2 Bases.
2. NEI 99-01 AA3 .

Category C – Cold Shutdown / Refueling System Malfunction

EAL Group: Cold Conditions: 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, DEF – Defueled).

The events of this category pertain to the following subcategories:

1. RPV Level

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

2. Loss of Critical AC Power

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

3. RCS Temperature

Uncontrolled or inadvertent temperature or pressure rises 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 125V DC vital 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.

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6. 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: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: UNPLANNED loss of RPV inventory

EAL:

CU1.1 Unusual Event

UNPLANNED loss of reactor coolant results in RPV water level lower than the prescribed level band for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is 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:

RPV level is normally monitored using the following instruments (ref. 1, 2):

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

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

CNS is equipped with multiple RPV water level instruments including: Wide Range, Fuel Zone, Shutdown Range, Steam Nozzle Range, and Narrow Range (ref. 1). Multiple instruments on different reference and variable legs should be monitored. The Upset Range and Shutdown Range instruments share a common reference leg; therefore, Narrow Range instruments should be routinely monitored when relying on Shutdown or Upset Range instrument as the primary indication.

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With the plant in Cold Shutdown, RPV water level is normally maintained above the RPV low level scram setpoint of 3.0 in. (ref. 3). However, if RPV level is being controlled below the RPV low level scram setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

With the plant in Refueling mode, RPV water level is normally maintained at or above the reactor vessel flange. Technical Specifications require at least 21 ft. of water above the top of the reactor vessel flange in the refueling cavity during refueling operations (ref. 4). The RPV flange is at approximately 206 in. on the Shutdown Range. (ref. 2).

This EAL 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.

Reference(s):

1. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
2. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
3. EOP-1A RPV Control.
4. Technical Specifications Section 3.9.7, RHR – High 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

EAL:

CU1.2 Unusual Event

RPV water level **cannot** be determined

AND EITHER

- UNPLANNED rise in any Table C-1 sump or pool level due to a loss of RPV inventory
- Visual observation of UNISOLABLE RCS leakage

Table C-1 Sumps/Pool
• Drywell equipment drain sump • Drywell floor drain sump • Reactor Building equipment drain sump • Reactor Building floor drain sump • Torus

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

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.

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Basis:

In this EAL, all water level indication is unavailable and the RPV inventory loss should be detected by the leakage indications listed in Table C-1. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV (ref. 1). A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling Mode, an unexplained rise in torus water level could be indicative of RHR valve misalignment or leakage (ref. 2). Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory.

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

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

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

This EAL 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.

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

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Reference(s):

1. System Operating Procedure 2.2.27, Equipment, Floor, and Chemical Drain System.
2. System Operating Procedure 2.2.69, Residual Heat Removal System.
3. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
4. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
5. NEI 99-01 CU1.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Significant Loss of RPV inventory

EAL:

CA1.1 Alert
Loss of RPV inventory as indicated by RPV water level < -42 in. (low-low)

Mode Applicability:

4 – Cold Shutdown, 5 – Refueling

Definition(s):

None

Basis:

The threshold RPV water level of -42 in. is the low-low ECCS actuation setpoint (Level 2). Although RCIC cannot restore RPV inventory in the cold condition, the Level 2 actuation setpoint is operationally significant and is indicative of a loss of RPV inventory significantly below the low RPV water level scram setpoint (ref. 1).

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

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

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

This EAL 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 RPV water level below the specified level 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.

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Although related, this EAL is concerned with the loss of RPV inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Residual Heat Removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

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

Reference(s):

1. Technical Specification Table 3.3.5.1-1.
2. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
3. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
4. NEI 99-01 CA1.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Significant Loss of RPV inventory

EAL:

CA1.2 Alert

RPV water level **cannot** be determined for ≥ 15 min. (Note 1)

AND EITHER

- UNPLANNED rise in **any** Table C-1 sump or pool level due to a loss of RPV inventory
- Visual observation of UNISOLABLE RCS leakage

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Table C-1 Sumps/Pool
• Drywell equipment drain sump • Drywell floor drain sump • Reactor Building equipment drain sump • Reactor Building floor drain sump • Torus

Mode Applicability:

4 – Cold Shutdown, 5 – Refueling

Definition(s):

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.

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Basis:

In this EAL, all water level indication is unavailable and the RPV inventory loss should be detected by the leakage indications listed in Table C-1. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV (ref. 1). A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling Mode, an unexplained rise in torus water level could be indicative of RHR valve misalignment or leakage (ref. 2). Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory.

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

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

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

This EAL 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 determine 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 determined, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RPV.

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

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

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Reference(s):

1. System Operating Procedure 2.2.27, Equipment, Floor, and Chemical Drain System..
2. System Operating Procedure 2.2.69, Residual Heat Removal System
3. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
4. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
5. 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 < -113 in. (low-low-low)

Mode Applicability:

4 – Cold Shutdown, 5 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Secondary Containment Closure requirements are defined in AP 0.50.5 Outage Shutdown Safety.

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:

The threshold RPV water level of -113 in. is the low-low-low ECCS actuation setpoint (Level 1). The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further lowering of RPV water level and potential core uncovery (ref. 1).

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

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

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

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

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

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 EALs 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 AG1.

Reference(s):

1. Technical Specifications Table 3.3.5.1-1, Emergency Core Cooling System Instrumentation.
2. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
3. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
4. NEI 99-01 CS1.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.2 Site Area Emergency

CONTAINMENT CLOSURE established

AND

No means of adequate core cooling can be assured, Table C-2

Table C-2 Adequate Core Cooling	
Core Submergence	RPV water level ≥ -158 in. (TAF)
Spray Cooling	Core spray flow > 4,750 gpm from a single subsystem **AND** RPV water level > -209 in.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Secondary Containment Closure requirements are defined in AP 0.50.5 Outage Shutdown Safety.

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:

Table C-2 specifies the means of adequate core cooling under cold shutdown and refueling conditions (ref. 1).

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When RPV level drops to the top of active fuel (TAF) (an indicated RPV level of -158 in.), core uncovery starts to occur (ref. 2). The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further lowering of RPV water level resulting in core uncovery.

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

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

Procedure 2.4RXLVL provides guidance for erratic or unexplained RPV water level changes. EOP/SAG Caution #1 indicates when an instrument may be used for level indication in the EOPs/SAGs (ref. 3).

4,721 gpm (rounded to 4750 gpm for readability) is the design core spray flow (ref. 1). This is the adequate core spray flow required to assure adequate core cooling, under cold conditions, provided that flow is obtained by a single core spray subsystem and RPV water level is above the elevation of the jet pump suction (-209 in.).

This EAL addresses a loss of ability to assure adequate core cooling due to a significant and prolonged loss of RPV level control and makeup capability in combination with insufficient core spray cooling 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 or adequate core spray flow cannot be established, 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 EALs 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.

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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 AG1.

Reference(s):

1. AMP-TBD00 PSTG/SATG Technical Basis.
2. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
3. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
4. NEI 99-01 CS1.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.3 Site Area Emergency

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

AND

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

- UNPLANNED rise in torus level of sufficient magnitude to indicate core uncovery
- Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Mode Applicability:

4 – Cold Shutdown, 5 – Refueling

Definition(s):

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.

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.

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Basis:

In this EAL, all water level indication is unavailable and the RPV inventory loss should be detected by leakage indications of a magnitude that may result in core uncovery. With RHR System operating in the Shutdown Cooling Mode, an unexplained rise in torus water level could be indicative of RHR valve misalignment or leakage (ref. 1). Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory.

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

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

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

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

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

In this EAL, 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.

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The inability to determine 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 determined, operators may determine that an inventory loss is occurring by observing changes in torus level or visual observation of RCS leakage of a magnitude that may result in core uncovery.

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 AG1.

Reference(s):

1. System Operating Procedure 2.2.69, Residual Heat Removal System.
2. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
3. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
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

No means of adequate core cooling can be assured, Table C-2 for ≥ 30 min. (Note 1)

AND

Any Containment Challenge indication, Table C-3

Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

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

Table C-2 Adequate Core Cooling	
Core Submergence	RPV water level ≥ -158 in. (TAF)
Spray Cooling	Core spray flow > 4,750 gpm from a single subsystem **AND** RPV water level > -209 in.

Table C-3 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • PC hydrogen concentration > 6% • UNPLANNED rise in PC pressure • Secondary Containment area radiation > 1,000 mR/hr (EOP-5A Table 10)

(continued on next page)

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Secondary Containment Closure requirements are defined in AP 0.50.5 Outage Shutdown Safety.

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.

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:

When RPV level drops to the top of active fuel (TAF) (an indicated RPV level of -158 in.), core uncovery starts to occur (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 lowering of RPV water level resulting in core uncovery.

RPV level is normally monitored using the following instruments (ref. 1, 2):

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

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

4,721 gpm (rounded to 4750 gpm for readability) is the design core spray flow. This is the adequate core spray flow required to assure adequate core cooling, under cold conditions, provided that flow is obtained by a single core spray subsystem and RPV water level is above the elevation of the jet pump suction (-209 in.) (ref. 3).

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Four conditions are associated with a challenge to Containment integrity:

- CONTAINMENT CLOSURE is not established.
- In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of 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 hydrogen concentration has exceeded the minimum necessary to support a hydrogen explosion – 6% (deflagration) (ref. 3).
- Any UNPLANNED rise in Primary Containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of CONTAINMENT CLOSURE capability. UNPLANNED Primary Containment pressure rise indicates CONTAINMENT CLOSURE cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.
- 1,000 mR/hr is the Secondary Containment Maximum Safe Operating radiation value. Exceeding this value is indicative of problems in the Secondary Containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-5A, Secondary Containment Control, Table 10. As indicated by NOTE 5 in EOP-5A, Table 10, RP Surveys and ARM Teledosimetry System may be used for these indications (ref. 4).

This EAL 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.

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*.

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Reference(s):

1. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
2. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
3. AMP-TBD00 PSTG/SATG Technical Basis.
4. EOP-5A, Secondary Containment Control, Table 10.
5. NEI 99-01 CG1.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CG1.2 General Emergency

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

AND

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

- UNPLANNED rise in torus level of sufficient magnitude to indicate core uncovery
- Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery

AND

Any Containment Challenge indication, Table C-3

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is 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-3 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • PC hydrogen concentration > 6% • UNPLANNED rise in PC pressure • Secondary Containment area radiation > 1,000 mR/hr (EOP-5A Table 10)

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

(continued on next page)

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Secondary Containment Closure requirements are defined in AP 0.50.5 Outage Shutdown Safety.

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.

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.

Basis:

In this EAL, all water level indication is unavailable and the RPV inventory loss should be detected by leakage indications of a magnitude that may result in core uncovery. With RHR System operating in the Shutdown Cooling Mode, an unexplained rise in torus water level could be indicative of RHR valve misalignment or leakage (ref. 1). Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory.

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

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

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

(continued on next page)

Four conditions are associated with a challenge to Containment integrity:

- CONTAINMENT CLOSURE is not established.
- In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of 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 hydrogen concentration has exceeded the minimum necessary to support a hydrogen explosion – 6% (deflagration) (ref. 4).
- Any UNPLANNED rise in Primary Containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of CONTAINMENT CLOSURE capability. UNPLANNED Primary Containment pressure rise indicates CONTAINMENT CLOSURE cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.
- 1,000 mR/hr is the Secondary Containment Maximum Safe Operating radiation value. Exceeding this value is indicative of problems in the Secondary Containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-5A, Secondary Containment Control, Table 10. As indicated by NOTE 5 in EOP-5A, Table 10, RP Surveys and ARM Teledosimetry System may be used for these indications (ref. 5).

This EAL 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.

(continued on next page)

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 determine 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 determined, operators may determine that an inventory loss is occurring by observing changes in torus level or visual observation of RCS leakage of a magnitude that may result in core uncovery.

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*.

Reference(s):

1. System Operating Procedure 2.2.69, Residual Heat Removal System.
2. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
3. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
4. AMP-TBD00, Step PC/H.
5. EOP-5A, Secondary Containment Control, Table 10.
6. NEI 99-01 CG1.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Critical AC Power

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

EAL:

CU2.1 Unusual Event

AC power capability, Table C-4, to critical 4160V buses 1F and 1G reduced to a single power source for ≥ 15 min. (Note 1)

AND

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Table C-4 AC Power Sources
Offsite • Startup Station Service Transformer • Emergency Station Service Transformer • Back-feed 345 kV line via Main Power Transformer to Normal Station Service Transformer (if already established) **Onsite** • DG-1 • DG-2

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, DEF - 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):

(continued on next page)

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:

This EAL 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 EOPs, and capable of supplying required power to a critical bus. Two examples of this condition are presented below:

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

The 4160V Critical Buses 1F (Div 1) and 1G (Div 2) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following off-site sources of power (ref. 1, 2, 3, 4):

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

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- Emergency Transformer - The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by either the 69 kV transmission line from OPPD or the Cooper 161 kV transformer T-6. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.
- Back-feeding power from the 345 kV line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on-line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. The time required to establish the back-feed is likely longer than the 15 minute interval. If off-normal plant conditions have already established the back-feed its power to the safety-related buses may be considered an off-site power source.

On-site power sources, under cold plant conditions, are the emergency diesel generators (DG-1 and DG-2) (ref. 5).

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 EAL CA2.1.

This EAL is the cold condition equivalent of the hot condition EAL SA1.1.

Reference(s):

1. System Operating Procedure 2.2.15, Startup Transformer.
2. System Operating Procedure 2.2.16, Normal Station Service Transformer.
3. System Operating Procedure 2.2.17, Emergency Station Service Transformer.
4. System Operating Procedure 2.2.18, 4160V Auxiliary Power Distribution System.
5. System Operating Procedure 2.2.20, Standby AC Power System (Diesel Generator).
6. NEI 99-01 CU2.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of ESF AC Power

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

EAL:

CA2.1 Alert
Loss of **all** offsite and **all** onsite AC power to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, DEF - 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:

This EAL 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.

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

The 4160V Critical Buses 1F (Div 1) and 1G (Div 2) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following off-site sources of power (ref. 1, 2, 3, 4):

- Startup Transformer - The Startup Transformer provides a source of off-site AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of off-site AC power to the station whenever the main generator is off-line. The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the station Normal Transformer fails or that the main generator trips off-line.
- Emergency Transformer - The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by either the 69 kV transmission line from OPPD or the Cooper 161 kV transformer T-6. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.
- Back-feeding power from the 345 kV line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on-line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. The time required to establish the back-feed is likely longer than the 15-minute interval. If off-normal plant conditions have already established the back-feed its power to the safety-related buses may be considered an off-site power source.

On-site power sources are the emergency diesel generators (DG-1 and DG-2) (ref. 5).

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Also under this EAL, provided the 15-minute interval can be met, credit can also be taken for non-safety-related power sources provided that operation of those sources is controlled in accordance with abnormal or emergency operating procedures (e.g., Supplemental Diesel Generator [SDG]) (ref. 6), or beyond design basis accident response guidelines (e.g., FLEX support guidelines) (ref. 7, 8). Such power sources should generally meet the "Alternate ac source" definition provided in 10CFR50.2.

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 AS1.

This EAL is the cold condition equivalent of the hot condition EAL SS1.1.

Reference(s):

1. System Operating Procedure 2.2.15, Startup Transformer.
2. System Operating Procedure 2.2.16, Normal Station Service Transformer.
3. System Operating Procedure 2.2.17, Emergency Station Service Transformer.
4. System Operating Procedure 2.2.18, 4160V Auxiliary Power Distribution System.
5. System Operating Procedure 2.2.20, Standby AC Power System (Diesel Generator).
6. EPFAQ 2015-015.
7. FLEX Support Guideline 5.10FLEX.07 4160V "F" Bus Tie-In with Off-Site Generator.
8. FLEX Support Guideline 5.10FLEX.08 4160V "G" Bus Tie-In with Off-Site Generator.
9. NEI 99-01 CA2.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED rise in RCS temperature

EAL:

CU3.1 Unusual Event
UNPLANNED rise in RCS temperature to ≥ 212°F

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Secondary Containment Closure requirements are defined in AP 0.50.5 Outage Shutdown Safety.

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:

Several instruments are capable of providing indication of RPV temperature with respect to the Technical Specification cold shutdown temperature limit (212°F) (ref. 1). These include (ref. 2, 3):

- NBI-TR-89, REACTOR VESSEL METAL TEMPERATURE RECORDER (Panel 9-21).
- Vessel Drain, PMIS Point M180, or NBI-TR-89 - Point 07 if M180 is not available.
- Vessel Bottom Head, PMIS Point M184, or NBI-TR-89 - Point 11 if M184 is not available.
- Bottom Head Adjacent to Support Skirt, PMIS Point M183, or NBI-TR-89 - Point 10.
- RR-TR-165, RR SUCTION & FEEDWATER TEMP (Panel 9-4).

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

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Inservice leak testing, hydrostatic testing, and control rod scram time testing in which RCS temperature is intentionally raised above 212°F per Technical Specification LCO 3.10.1 are not applicable to this EAL (ref. 4).

This EAL addresses an UNPLANNED increase in RCS temperature greater than or equal to 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 EAL CA3.1.

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 a refueling 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.

Reference(s):

1. Technical Specifications Table 1.1-1.
2. General Operating Procedure 2.1.1, Startup Procedure.
3. System Operating Procedure 2.2.69.2, RHR System Shutdown Operations.
4. Technical Specifications LCO 3.10.1.
5. NEI 99-01 CU3.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED rise in RCS temperature

EAL:

CU3.2 Unusual Event

Loss of **all** RCS temperature and RPV water level indication for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5- Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Secondary Containment Closure requirements are defined in AP 0.50.5 Outage Shutdown Safety.

Basis:

This EAL addresses the inability to determine RCS temperature and level, and represents a potential degradation of the level of safety of the plant.

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.

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RPV level is normally monitored using the following instruments (ref.1, 2):

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

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

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

Several instruments are capable of providing indication of RPV temperature with respect to the Technical Specification cold shutdown temperature limit (212°F) (ref. 5). These include (ref. 3, 4):

- NBI-TR-89, REACTOR VESSEL METAL TEMPERATURE RECORDER (Panel 9-21).
- Vessel Drain, PMIS Point M180, or NBI-TR-89 - Point 07 if M180 is not available.
- Vessel Bottom Head, PMIS Point M184, or NBI-TR-89 - Point 11 if M184 is not available.
- Bottom Head Adjacent to Support Skirt, PMIS Point M183, or NBI-TR-89 - Point 10.
- RR-TR-165, RR SUCTION & FEEDWATER TEMP (Panel 9-4).

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

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.

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Reference(s):

1. Abnormal Procedure 2.4RXLVL, RPV Water Level Control Trouble.
2. Instrument Operating Procedure 4.6.1, Reactor Vessel Water Level Indication.
3. General Operating Procedure 2.1.1, Startup Procedure.
4. System Operating Procedure 2.2.69.2, RHR System Shutdown Operations.
5. Technical Specifications Table 1.1-1.
6. NEI 99-01 CU3.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.1 Alert

UNPLANNED rise in RCS temperature to ≥ 212°F for > Table C-5 duration (Note 1)

OR

UNPLANNED RPV pressure rise > 10 psig

Note 1: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Table C-5 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 actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Secondary Containment Closure requirements are defined in AP 0.50.5 Outage Shutdown Safety.

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).

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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:

In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the RCS pressure rise criteria when the RCS is intact in Mode 4 or based on time to boil data when in Mode 5 or the RCS is not intact in Mode 4 (ref. 7).

This EAL 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 (ref. 6) 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 containment atmosphere and subsequently to the environment, and 2) there is reduced reactor coolant inventory above the top of irradiated fuel.

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

A 10 psig RPV pressure increase can be read on (ref. 1, 2):

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

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Several instruments are capable of providing indication of RPV temperature with respect to the Technical Specification cold shutdown temperature limit (212°F) (ref. 6). These include (ref. 3, 4):

- NBI-TR-89, REACTOR VESSEL METAL TEMPERATURE RECORDER (Panel 9-21).
- Vessel Drain, PMIS Point M180, or NBI-TR-89 - Point 07 if M180 is not available.
- Vessel Bottom Head, PMIS Point M184, or NBI-TR-89 - Point 11 if M184 is not available.
- Bottom Head Adjacent to Support Skirt, PMIS Point M183, or NBI-TR-89 - Point 10.
- RR-TR-165, RR SUCTION & FEEDWATER TEMP (Panel 9-4).

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

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

Escalation of the emergency classification level would be via IC CS1 or AS1.

Reference(s):

1. Instrument Operating Procedure 4.6.2, Reactor Vessel Pressure Indication.
2. Emergency Procedure 5.9SAMG, Severe Accident Management Guidance, Technical Support Guidelines attachment (CPA TSG).
3 General Operating Procedure 2.1.1, Startup Procedure.
4. System Operating Procedure 2.2.69.2, RHR System Shutdown Operations.
5. Technical Specifications LCO 3.10.1.
6. Technical Specifications Table 1.1-1.
7. AP 2.4SDC Shutdown Cooling Abnormal, Attachment 5, Time to Core Boiling/Time to Core Uncovery.
8. 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
Indicated voltage is < 105 VDC on **required** Technical Specification vital 125 VDC bus 1A or 1B for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

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:

This EAL 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.

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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 as required by Technical Specifications (ref. 1). For example, if Train A is out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an Unusual Event. A loss of Vital DC power to Train A would not warrant an emergency classification.

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

105 VDC is the minimum design bus voltage (ref. 2, 5).

The 125 VDC System supplies DC power to conventional station emergency equipment and selected Safeguard System loads. 125 VDC Distribution Panels supply control and instrument power for annunciators, control logic power and protective relaying.

If 125 VDC Distribution Panel A is lost, the following major equipment is affected: RRMG A speed and breaker control, 4160V Bus 1A, 1E, and 1F breaker control and undervoltage logics, 480V Bus 1A and 1F breaker control, the right light in all Control Room annunciators, annunciator panels for Water Treatment, RHR A Gland Water, Auxiliary Steam Boiler C, DG-1 starting and breaker control logics, CS A, RCIC, and RHR A control logics, TIP valve control monitors, main generator voltage regulation, RFPT A trip logic, and ARI solenoid valve power (ref. 5).

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

Battery chargers receive their power from 460V critical motor control centers. Each 125 VDC bus receives power from either a 125 VDC battery or a 125 VDC battery charger. The 250 VDC System supplies DC power to conventional station emergency equipment and selected Safeguard System loads. Although 250 VDC Buses 1A and 1B provide vital DC emergency power, 250 VDC Safety System loads (such as motor operated valves) also require 125 VDC control power. Loss of 125 VDC buses alone, therefore, would render most Safeguard System loads inoperable (ref. 3, 4, 5).

Depending upon the event, escalation of the emergency classification level would be via IC CA1 or CA3, or an IC in Category A.

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

(continued on next page)

Reference(s):

1. Technical Specifications 3.8.5 DC Sources - Shutdown.
2. Technical Specifications B 3.8.4 DC Sources - Operating.
3. USAR Section VIII-6.2.
4. USAR Section VIII-6.3.
5. Emergency Procedure 5.3DC125, Loss of 125 VDC.
6. 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** communication methods, Table C-6, for **any** of the following:

- Onsite
- State/Local
- NRC

Table C-6 Communication Methods

System	Onsite	State/Local	NRC
Station Intercom System (Gaitronics)	X		
Site UHF Radio Consoles	X		
Alternate Intercom	X		
CNS On-Site Cell Phone System	X	X	X
Telephone System (PBX)	X	X	X
Local Telephones (C. O. Lines)		X	X
Satellite Telephones		X	X
CNS State Notification Telephones		X	
Federal Telecommunications System (FTS 2001)			X

Mode Applicability:

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

(continued on next page)

Definition(s):

None

Basis:

This EAL addresses a significant loss of on-site or offsite communications capabilities (ref. 1, 2). While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to State and local agencies and the NRC.

This EAL 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 bullet addresses a total loss of the communications methods used in support of routine plant operations.

The second bullet addresses a total loss of the communications methods used to notify all State and local agencies of an emergency declaration. The State and local agencies referred to here are the Nebraska State Patrol, Missouri State Patrol, Atchison County Sheriff's Department and the Nemaha and Richardson County Sheriff's Departments.

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

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

Reference(s):

1. EPIP 5.7COMMUN, Communications, Emergency Response Facility Communication Equipment attachment.
2. Emergency Plan for Cooper Nuclear Station, Section 7.3 Communications Systems and Notification.
3. NEI 99-01 CU5.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting Safety Systems

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

EAL:

CA6.1 Alert

The occurrence of **any** Table C-7 hazardous event

AND

Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode

AND EITHER:

- Event damage has caused indications of degraded performance to the redundant train of the SAFETY SYSTEM needed for the current operating mode
- Event damage has resulted in VISIBLE DAMAGE to the redundant train of the SAFETY SYSTEM needed for the current operating mode

(Notes 9, 10)

Note 9: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then emergency classification is **not** warranted.

Note 10: If the hazardous event **only** resulted in VISIBLE DAMAGE, with **no** indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is **not** warranted.

Table C-7 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

(continued on next page)

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 SAFETY SYSTEM train 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 SAFETY SYSTEM train.

(continued on next page)

Basis:

This EAL addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the redundant train such that the potential exists for this redundant SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second (redundant) SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance 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.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. 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. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via IC CS1 or AS1.

This EAL is the cold condition equivalent of the hot condition EAL SA8.1.

Reference(s):

1. EP FAQ 2016-002.
2. NEI 99-01 CA6.

Category E – Independent Spent Fuel Storage Installation (ISFSI)

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

An independent spent fuel storage installation (ISFSI) is a complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage. A significant amount of the radioactive material contained within a 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.

An Unusual Event is declared on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated.

The CNS ISFSI is located wholly within the plant PROTECTED AREA. Therefore, any security event related to the ISFSI is classified under Category H1 security event related EALs.

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 on the surface of a loaded spent fuel cask (HSM) > **any** of the following:

- 1,300 mrem/hr at the HSM bird screen
- 4 mrem/hr outside HSM door
- 8 mrem/hr on end shield wall exterior

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY - The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the CNS ISFSI, Confinement Boundary is defined as the NUHOMS Dry Shielded Canister (DSC).

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.

Basis:

This EAL 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.

(continued on next page)

The existence of "damage" is determined by radiological survey. The specified EAL threshold values correspond to 2 times the most limiting cask design technical specification values. The technical specification (licensing bases document) multiple of "2 times", which is also used in Category A IC AU1, is used here to distinguish between non-emergency and emergency conditions (ref. 2). 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.

The dry-cask storage system is the NUHOMS Horizontal Modular Storage System. (ref. 1). CNS uses a combination of NUHOMS Dry Shielded Canister (DSC) models 61BT (Standardized Horizontal Storage Module [HSM]) and 61BTH (HSM-H).

The values shown represent 2 times the most limiting limits specified in the ISFSI Certificate of Compliance Technical Specification section 5.4.2 for radiation external to an HSM loaded with a model 61BTH DSC (ref. 1, 2).

Security-related events for ISFSIs are covered under EALs HU1.1 and HA1.1.

Reference(s):

1. Technical Specifications for the Standardized NUHOMS Horizontal Modular Storage System, Certificate of Compliance 1004 Amendment No. 13 Revision 1 Section 5.4.
2. Emergency Procedure 5.1HSM, Horizontal Storage Module Integrity.
3. NEI 99-01 E-HU1.

Category F – Fission Product Barrier Degradation©[4]

EAL Group: Hot Conditions; EALs in this category are applicable only in one or more hot operating modes.

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

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

B. Reactor Coolant System (RCS): The RCS barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping out to and including the isolation valves.

C. Primary Containment (PC): The Primary Containment barrier includes the drywell, the wetwell (torus), their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. 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.

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. "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Alert:

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

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The logic used for emergency classification based on fission product barrier monitoring should reflect the following considerations:

- The Fuel Clad barrier and the RCS Barrier are weighted more heavily than the Primary Containment barrier.
- Unusual Event ICs associated with RCS and Fuel Clad barriers are addressed under System Malfunction ICs.
- For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC AG1 has been exceeded.
- The fission product barrier thresholds specified within a scheme reflect plant-specific CNS 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 containment, an interfacing system, or outside of the 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 barrier

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:

Fuel Clad, RCS and Primary Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds.

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.

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):

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:

Fuel Clad, RCS and Primary Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds.

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.

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:

Fuel Clad, RCS and Primary Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds.

At the General Emergency classification level not every 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

Reference(s):

1. NEI 99-01 FG1.

Table F-1 Fission Product Barrier Threshold Matrix & Bases

Table F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Primary Containment). The table is structured so that each of the three barriers occupies adjacent columns. Each fission product barrier column is further divided into two columns; one for Loss thresholds and one for Potential Loss thresholds.

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

A. RPV Water Level

B. RCS Leak Rate

C. Containment Conditions

D. Primary Containment Radiation / RCS Activity

E. Primary Containment Integrity or Bypass

F. Emergency Director Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word "None" is entered in the cell.

Thresholds are assigned sequential numbers within each barrier column beginning with number one (ex., FC1, FC2...FC6).

If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of Table F-1, locates the likely category and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine if a threshold has been exceeded. If a threshold has not been exceeded, the EAL-user proceeds to the next likely category and continues review of the thresholds in the new category

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If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded, only that one barrier is lost or potentially lost. The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if containment radiation is sufficiently high, a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Primary Containment barrier can occur. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, and FA1.1 to determine the appropriate emergency classification.

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

Table F-1 Fission Product Barrier Threshold Matrix©[4]

	Fuel Clad Barrier (FC)		Reactor Coolant System Barrier (RCS)		Primary Containment Barrier (PC)	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A **RPV Water Level**	FC1 SAG 1 entry is required	FC2 RPV water level **cannot** be restored and maintained > -158 in. (TAF) or **cannot** be determined	RCS1 RPV water level **cannot** be restored and maintained > -158 in. (TAF) or **cannot** be determined	None	None	PC1 It **cannot** be determined that core debris will be retained in the RPV
B **RCS Leak Rate**	None	None	RCS2 UNISOLABLE break outside PC in **any** of the following: • Main steam line • HPCI steam line • RCIC steam Line • RWCU • Feedwater RCS3 Emergency RPV Depressurization is required	RCS4 UNISOLABLE primary system leakage that results in exceeding **EITHER**: • One or more secondary containment Maximum Normal Operating temperature limit (EOP-5A Table 9) • One or more secondary containment Maximum Normal Operating radiation limit (EOP-5A Table 10)	PC2 UNISOLABLE primary system leakage that results in exceeding **EITHER**: • One or more secondary containment Maximum Safe Operating temperature limit (EOP-5A Table 9) • One or more secondary containment Maximum Safe Operating radiation limit (EOP-5A Table 10)	None
C **PC Conditions**	None	None	RCS5 Drywell pressure > 1.84 psig due to RCS leakage	None	PC3 UNPLANNED rapid drop in PC pressure following PC pressure rise PC4 PC pressure response **not** consistent with LOCA conditions	PC5 PC pressure > 56 psig PC6 Deflagration concentrations exist inside PC (≥ 6% H_2 and ≥ 5% O_2) PC7 HCTL curve exceeded (EOP/SAG Graph 7)
D **PC Rad / RCS Activity**	FC3 Drywell radiation monitor (RMA-RM-40A/B) ≥ 3.6E+3 Rem/hr FC4 Primary coolant activity > 300 µCi/gm dose equivalent I-131	None	RCS6 Drywell radiation monitor (RMA-RM-40A/B) ≥ 49 Rem/hr	None	None	PC8 Drywell radiation monitor (RMA-RM-40A/B) ≥ 1.9E+4 Rem/hr
E **PC Integrity or Bypass**	None	None	None	None	PC9 UNISOLABLE direct downstream pathway to the environment exists after PC isolation signal PC10 Intentional PC venting per EOPs/SAGs irrespective of offsite radioactivity release rates	None
F **Emergency Director Judgment**	FC5 **Any** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier	FC6 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier	RCS7 **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	RCS8 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	PC11 **Any** condition in the opinion of the Emergency Director that indicates loss of the PC barrier	PC12 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the PC barrier

Barrier: Fuel Clad

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

FC1
SAG 1 entry is required

Definition(s):

None

Basis:

This Fuel Clad Loss threshold is indicative of a severe loss of adequate core cooling.

EOP-1A, EOP-1B, EOP-2B, EOP-7A, and EOP-7B specify entry into SAG 1 when it is determined that core damage is occurring due to loss of core cooling (Reference 1). SAG entry signifies the need to implement severe accident mitigation actions. These EOPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined.

SAG 1 entry is required during Non-Failure-to-Scram events when any of the following conditions exist (Reference 1):

- RPV water level cannot be restored and maintained above -183 inches (MSCRWL, EOP-1A) (Reference 2, 7).
- RPV water level cannot be restored and maintained at or above -209 inches (elevation of the jet pump suction) and no core spray subsystem flow can be restored and maintained equal to or ≥ 4,750 gpm (design core spray flow, EOP-1A) (Reference 2, 9).

SAG 1 entry is required during Failure-to-Scram events when RPV water level cannot be restored and maintained > -183 inches and core steam flow cannot be restored and maintained > 800,000 lbm/hr (Reference 3, 7). The specified steam flow is the Minimum Core Steam Flow (MCSF).

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The MCSF is the lowest core steam flow sufficient to preclude any clad temperature from exceeding 1500°F even if the reactor core is not completely covered (Reference 6). The MCSF is only applicable in failure-to-scram events because reactor power must be well above the decay heat generation rate for steam production to equal the MCSF.

Whether or not RPV water level can be determined, SAG 1 entry is also required when core damage is occurring due to loss of core cooling (EOP-1A, EOP-1B, EOP-2B, EOP-7A, or EOP-7B) (Reference 2, 3, 4, 5, 6). If RPV water level cannot be determined, the absence of core damage indications may be the only means of determining if adequate core cooling is being maintained. If RPV water level can be determined, restoration of RPV water level to above -183 inches and restoration of core spray cooling requirements may not occur in a timely manner. If indications of core damage occur while RPV injection is being restored, entry to SAG 1 is appropriate even if the required water levels and spray cooling flow are eventually achieved.

The above EOP conditions represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

This threshold has been aligned to the updated guidance of EPG/SAG Revision 4 such that the plant Operator can readily implement both when needed (ref. 9, 10).

Reference(s):

1. AMP-TBD00 PSTG/SATG Technical Bases, RC/L, Contingency #1, #4, #5.
2. EOP-1A, RPV Control.
3. EOP-7A, RPV Level (Failure-to-Scram).
4. EOP-2B, RPV Flooding.
5. EOP-7B, RPV Flooding (Failure-to-Scram).
6. EOP-1B, Alternate Level/Pressure Control.
7. NEDC 97-090J.
8. NEDC 97-089.
9. EP FAQ 2015-004.
10. BWROG EPG/SAG Revision 4.
11. NEI 99-01, RPV Water Level Fuel Clad Loss 2.A.

Barrier: Fuel Clad

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

FC2

RPV water level **cannot** be restored and maintained > -158 in. (TAF) or **cannot** be determined

Definition(s):

None

Basis:

This water level (-158 in.) corresponds to the top of the active fuel and is used in the EOPs to indicate a challenge to core cooling (ref. 1). When RPV level is at or above the TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV water level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops to TAF, the level is indicative of a challenge to core cooling and the Fuel Clad barrier.

When RPV water level cannot be determined, EOPs require entry to RPV Flooding procedures. 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 the RPV Flooding procedures 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 (ref. 2, 4). If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the Fuel Clad barrier exists.

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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, EALs SA6.1 or SS6.1 will dictate the need for emergency classification (ref. 3).

The RPV water level threshold is the same as RCS barrier Loss threshold RCS1. 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 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.

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 (ref. 2, 4).

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Reference(s):

1. EOP-1A, RPV Control.
2. EOP-2B, RPV Flooding.
3. EOP-7A, RPV Level (Failure-to-Scram).
4. EOP-7B, RPV Flooding (Failure-to-Scram).
5. NEI 99-01, RPV Water Level Potential Loss 2.A.

Barrier: Fuel Clad

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

FC3
Drywell radiation monitor (RMA-RM-40A/B) ≥ 3.6E+3 Rem/hr

Definition(s):

None

Basis:

The drywell radiation monitor reading (3.68E+3 Rem/hr rounded to 3.6E+3 Rem/hr for readability) corresponds to an instantaneous release of all reactor coolant mass into the 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 3.81% fuel clad damage (ref. 1). 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 RCS6 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 ECL to a Site Area Emergency.

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

Reference(s):

1. Calculation No. 19-011 EAL Rev. 6 Threshold Calculations (EP-CALC-CNS-1902 Containment Radiation EAL Threshold Values).
2. NEI 99-01, Primary Containment Radiation Fuel Clad Loss 4.A.

Barrier: Fuel Clad

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

FC4
Primary coolant activity > 300 µCi/gm dose equivalent I-131©[4]

Definition(s):

None

Basis:

This threshold indicates that RCS radioactivity concentration is greater than 300 µCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 1% fuel clad damage (ref. 1). Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications.

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

Reference(s):

1. NEDC 02-20, Estimation of Reactor Coolant System Dose Equivalent I-131 Concentration Following a 1% Fuel Clad Failure (Degraded Core) Under Non-LOCA Conditions.
2. NEI 99-01, RCS Activity Fuel Clad Loss 1.A.

Barrier: Fuel Clad

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

FC5 **Any** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad Barrier

Definition(s):

None

Basis:

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

Reference(s):

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

Barrier: Fuel Clad

Category: F. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

FC6

Any condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad Barrier

Definition(s):

None

Basis:

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

Reference(s):

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

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

RCS1

RPV water level **cannot** be restored and maintained > -158 in. (TAF) or **cannot** be determined

Definition(s):

None.

Basis:

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, EALs SA6.1 or SS6.1 will dictate the need for emergency classification (ref. 3).

This water level (-158 in.) corresponds to the top of active fuel and is used in the EOPs to indicate a challenge to core cooling (ref. 1). When RPV level is at or above the TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV water level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops to TAF, the level is indicative of a challenge to core cooling and the Fuel Clad barrier.

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When RPV water level cannot be determined, EOPs require entry to RPV Flooding procedures. 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 the RPV Flooding procedures 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 (ref. 2, 4). If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the Fuel Clad barrier exists.

The RPV water level threshold is the same as Fuel Clad barrier Potential Loss threshold FC2. 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 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.

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 beyond the limit**; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

There is no RCS Potential Loss threshold associated with RPV Water Level.

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Reference(s):

1. EOP-1A, RPV Control.
2. EOP-2B, RPV Flooding
3. EOP-7A, RPV Level (Failure-to-Scram).
4. EOP-7B, RPV Flooding (Failure-to-Scram).
5. NEI 99-01, RPV Water Level RCS Loss 2.A.

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

RCS2

UNISOLABLE break outside PC 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:

Failure to isolate the leak, within 15 minutes or if known that the leak cannot be isolated
within 15 minutes, from the start of the leak requires immediate classification.

The conditions of this threshold include required containment isolation failures allowing a flow path to the environment. A release pathway outside containment exists when flow is not prevented by downstream isolations. In the case of a failure of both isolation valves to close but in which no downstream flowpath exists, emergency declaration under this threshold would not be required. Similarly, if the emergency response requires the normal process flow of a system outside 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 <u>not</u> met.

This threshold condition represents the loss of both the RCS and PC (see Loss PC9) barriers and justifies declaration of a Site Area Emergency (i.e., Loss or Potential Loss of any two barriers).

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Even though RWCU and Feedwater systems do not contain steam, they are included in the list because an UNISOLABLE break could result in the high-pressure discharge of fluid that is flashed to steam from relatively large volume systems directly connected to the RCS.

Large high-energy lines that rupture outside PC 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, remotely or locally, the RCS barrier Loss threshold is met.

Reference(s):

1. NEI 99-01, RCS Leak Rate RCS Loss 3.A.

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

RCS3
Emergency RPV Depressurization is required

Definition(s):

None

Basis:

Emergency RPV Depressurization in accordance with the EOPs (ref. 1, 2) 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). Even though the RCS is being vented into the torus, a Loss of the RCS barrier exists due to the diminished effectiveness of the RCS to retain fission products within its boundary.

EOP-2A RPV Control - Emergency RPV Depressurization/Steam Cooling allows terminating the depressurization if necessary to maintain RCIC as an injection source. This would require closing the SRVs. Even though the SRVs may be reclosed, this threshold is still met due to the requirement for an Emergency Depressurization having been met (ref. 2).

Reference(s):

1. EOP-2A, Emergency RPV Depressurization/Steam Cooling.
2. EP FAQ 2015-003.
3. NEI 99-01, RCS Leak Rate RCS Loss 3.B.

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

RCS4

UNISOLABLE primary system leakage that results in exceeding **EITHER**:

- One or more secondary containment Maximum Normal Operating temperature limit (EOP-5A Table 9)
- One or more secondary containment Maximum Normal Operating radiation limit (EOP-5A Table 10)

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.

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

Basis:

Failure to isolate the leak, within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

The presence of elevated general area temperatures or radiation levels in the Secondary Containment may be indicative of UNISOLABLE primary system leakage outside the containment. The EOP-5A entry condition values define this RCS threshold because they are the maximum normal operating values and signify the onset of abnormal system operation. When parameters reach this level, equipment failure or mis-operation may be occurring. Elevated parameters may also adversely affect the ability to gain access to or operate equipment within the affected area. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-5A, Secondary Containment Control (ref. 1).

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In general, multiple indications should be used to determine if a primary system is discharging outside PC. 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.

Potential loss of RCS based on primary system leakage outside the PC is determined from EOP temperature or radiation EOP-5A 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 PC.

A Max Normal EOP-5A 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 EOP-5A Max Normal Operating values escalates to a Site Area Emergency when combined with Containment Barrier Loss thresholds PC2 or PC9 (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

Reference(s):

1. EOP-5A, Secondary Containment Control.
2. NEI 99-01, RCS Leak Rate RCS Potential Loss 3.A.

Barrier: Reactor Coolant System

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

RCS5
Drywell pressure > 1.84 psig due to RCS leakage

Definition(s):

None

Basis:

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

In the CNS design basis, 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 containment vent/purge (ref. 3).

The threshold phrase "...due to RCS leakage" focuses the barrier failure on the RCS instead of the non-LOCA malfunctions that may adversely affect containment pressure. Drywell pressure greater than 1.84 psig with corollary indications (e.g., drywell temperature, indications of loss of RCS inventory) should therefore be considered a Loss of the RCS barrier. Loss of drywell cooling that results in pressure greater than 1.84 psig should not be considered an RCS barrier Loss.

The 1.84 psig value 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 PC pressure.

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Reference(s):

1. EOP-1A, RPV Control.
2. EOP-3A, Primary Containment Control.
3. USAR Section XIV-6.3.
4. NEI 99-01, Primary Containment Pressure RCS Loss 1.A.

Barrier: Reactor Coolant System

Category: D. PC Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

RCS6
Drywell radiation monitor (RMA-RM-40A/B) ≥ 49 Rem/hr

Definition(s):

None

Basis:

The drywell radiation monitor reading (49.1 Rem/hr rounded to 49 Rem/hr for readibility) corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification 3.4.6 allowable limit of 4 μCi/gm dose equivalent I-131 (ref. 1). This value is lower than that specified for Fuel Clad Barrier Loss threshold FC3 since it indicates a loss of the RCS Barrier only.

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

Reference(s):

1. Calculation No. 19-011 EAL Rev. 6 Threshold Calculations (EP-CALC-CNS-1902 Containment Radiation EAL Threshold Values).
2. NEI 99-01, Primary Containment Radiation RCS Loss 4.A.

Barrier: Reactor Coolant System

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

RCS7 **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS Barrier

Definition(s):

None

Basis:

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

Reference(s):

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

Barrier: Reactor Coolant System

Category: F. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

RCS8
Any condition in the opinion of the Emergency Director that indicates potential loss of the RCS Barrier

Definition(s):

None

Basis:

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.

Reference(s):

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

Barrier: Primary Containment

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

PC1
It **cannot** be determined that core debris will be retained in the RPV

Definition(s):

None

Basis:

The determination that "It cannot be determined that core debris will be retained in the RPV" is made from the evaluation of criteria identified in the SAGs and the supporting Technical Support Guidelines (ref. 1, 2). If it cannot be determined that core debris will be retained in the RPV, then subsequent events could lead to a potential challenge to primary containment integrity. This decision would occur prior to RPV failure and the release of core debris into the primary containment.

This threshold will promote an accurate and timely General Emergency declaration, and preclude an unwarranted evacuation of the public. 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.

There is no PC loss threshold associated with RPV Water Level.

Reference(s):

1. SAG 2 Severe Accident Procedure 5.9SAMG Attachment 1.
2. TSG-3.16 Core Debris Can Be Retained in the RPV.
3. EP FAQ 2015-004.
4. NEI 99-01, RPV Water Level PC Potential Loss 2.A.

Barrier: Primary Containment

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

PC2

UNISOLABLE primary system leakage that results in exceeding **EITHER**:

- One or more secondary containment Maximum Safe Operating temperature limit (EOP-5A Table 9)
- One or more secondary containment Maximum Safe Operating radiation limit (EOP-5A Table 10)

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.

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

Basis:

Failure to isolate the leak, within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

The presence of elevated general area temperatures or radiation levels in the Secondary Containment may be indicative of UNISOLABLE primary system leakage outside the containment. The MAX SAFE values define this PC barrier threshold because they are indicative of problems in the Secondary Containment that are spreading and pose a threat to achieving a safe plant shutdown. This threshold addresses problematic discharges outside containment that may not originate from a high-energy line break. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-5A, Secondary Containment Control (ref. 1).

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In general, multiple indications should be used to determine if a primary system is discharging outside PC. For example, a high area temperature condition does not necessarily indicate that a primary system is discharging into the Secondary Containment since this may be caused by a fire or loss of area cooling. Conversely, a high area temperature condition in conjunction with other indications (e.g., room FLOODING, high area radiation levels, 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.

The Max Safe Operating area temperature values and the Max Safe Operating area radiation values 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.

There is no containment potential loss threshold associated with RCS Leak Rate.

Reference(s):

1. EOP-5A, Secondary Containment Control.
2. NEI 99-01, RCS Leak Rate PC Loss 3.C.

Barrier: Primary Containment

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

PC3
UNPLANNED rapid drop in PC pressure following PC 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:

Rapid UNPLANNED loss of PC pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure increase indicates a loss of PC 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.

Reference(s):

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

Barrier: Primary Containment

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

PC4
PC pressure response **not** consistent with LOCA conditions

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:

PC pressure should increase as a result of mass and energy release into the PC from a LOCA (ref. 1). Thus, PC pressure not increasing under these conditions indicates a loss of PC integrity.

These thresholds rely 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.

Reference(s):

1. USAR Section XIV-6.3.7.
2. NEI 99-01, Primary Containment Conditions PC Loss 1.B.

Barrier: Primary Containment

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

PC5
PC pressure > 56 psig

Definition(s):

None

Basis:

When the PC pressure exceeds the design value (56 psig) (ref. 1), containment venting may be required even if offsite radioactivity release rate limits will be exceeded (ref. 2). If this threshold is exceeded, a challenge to the containment structure has occurred because assumptions used in the accident analysis are no longer VALID and an unanalyzed condition exists. This constitutes a Potential Loss of the Containment barrier even if a containment breach has not occurred.

The threshold pressure is the PC internal design pressure. Structural acceptance testing demonstrates the capability of the PC 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 PC barrier.

Reference(s):

1. USAR Table V-2-1.
2. EOP-3A, Primary Containment Control.
3. NEI 99-01, Primary Containment Conditions PC Potential Loss 1.A.

Barrier: Primary Containment

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

PC6

Deflagration concentrations exist inside PC (≥ 6% H_2 and ≥ 5% O_2)

Definition(s):

None

Basis:

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 (deflagration) mixture of dissolved gases in the containment. However, containment monitoring and/or sampling should be performed to verify this assumption and to determined if hydrogen concentration has exceeded the minimum necessary to support a hydrogen deflagration (6%) in the presence of > 5% oxygen in either the drywell or torus (ref. 1, 2). Except for brief periods during plant startup and shutdown, oxygen concentration in the Primary Containment is maintained at insignificant levels by nitrogen inertion.

If hydrogen concentration reaches or exceeds the lower deflagration limit, as defined in plant EOPs, in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside the containment, loss of the Containment barrier could occur.

Drywell and suppression chamber atmosphere is monitored for H_2 and O_2 by a divisionally separated H_2/O_2 Monitoring System. The system consists of two H_2/O_2 analyzers (PC-AN-H2/O2I and PC-AN-H2/O2II), two remote process panels (PC-CS-H2/O2I and PC-CS-H2/O2II), two H_2 recorders (PC-R-H2I and PC-R-H2II), two O_2 recorders (PC-R-O2I and PC-R-O2II), an O_2 digital indicator (PC-I-1), associated control switches and sample stream indicating lights. H_2/O_2 analyzers are located in the Reactor Building at 976', remote process panels are located in the Cable Spreading Room, recorders are located on VBD-P1 and VBD-P2, the O_2 digital indicator and sample stream lights are located on VBD-H. Div 2 is normally in service providing O_2 concentration on VBD-H and H_2 and O_2 concentrations on PMIS (Reference 3).

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Reference(s):

1. EOP-3A, Primary Containment Control.
2. BWROG EPG/SAG Revision 2, Sections PC/G.
3. System Operating Procedure 2.2.60.1, Containment H2/O2 Monitoring System.
4. NEI 99-01, Primary Containment Conditions PC Potential Loss 1.B.

Barrier: Primary Containment

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

PC7 HCTL curve exceeded (EOP/SAG Graph 7)

Definition(s):

None

Basis:

The Heat Capacity Temperature Limit (HCTL) 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 HCTL 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.

Reference(s):

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

Barrier: Primary Containment

Category: D. PC Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

PC8
Drywell radiation monitor (RMA-RM-40A/B) ≥ 1.9E+4 Rem/hr

Definition(s):

None

Basis:

In order to reach this Containment barrier Potential Loss threshold, a loss of the RCS barrier (Loss RCS6) and a loss of the Fuel Clad barrier (Loss FC3) have already occurred. This threshold, therefore, represents a General Emergency classification.

The radiation monitor reading (1.94E+4 Rem/hr rounded to 1.9E+4 Rem/hr for readability) corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that 20% of the fuel cladding has failed (ref. 1). 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.

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

Reference(s):

1. Calculation No. 19-011 EAL Rev. 6 Threshold Calculations (EP-CALC-CNS-1902 Containment Radiation EAL Threshold Values).
2. NEI 99-01, Primary Containment Radiation Potential Loss 4.A.

Barrier: Primary Containment

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

PC9
UNISOLABLE direct downstream pathway to the environment exists after PC isolation signal

Definition(s):

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

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.

Basis:

Failure to isolate the leak, within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway directly to the environment. The concern is the UNISOLABLE open pathway to the environment. A failure of the ability to isolate any one line following a valid isolation signal indicates a breach of containment integrity.

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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). Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include UNISOLABLE main steam line or RCIC steam line breaks, UNISOLABLE RWCU system breaks, and UNISOLABLE containment atmosphere vent paths. However, if the main condenser is available with an UNISOLABLE main steam line, there may be releases through the steam jet air ejectors and gland seal exhausters. These pathways are monitored, however, and do not meet the intent of a non-isolable release path to the environment. These minor releases are assessed using the Category A, Abnormal Rad Release / Rad Effluent, EALs.

The 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.

EOP-3A Primary Containment Control, may specify containment venting and intentional bypassing of the PC isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1). Under these conditions with a VALID containment isolation signal, the PC barrier should be considered lost.

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

There is no PC Potential Loss threshold associated with PC Integrity or Bypass.

Reference(s):

1. EOP-3A, Primary Containment Control.
2. NEI 99-01, Primary Containment Isolation Failure PC Loss 3.A.

Barrier: Primary Containment

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

PC10
Intentional PC venting per EOPs/SAGs irrespective of offsite radioactivity release rates

Definition(s):

None

Basis:

EOP-3A, Primary Containment Control, may specify containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1). The threshold is met when the operator begins venting the containment in accordance with EOP-3A, SAG 1 or SAG 2, not when actions are taken to bypass interlocks prior to opening the vent valves. Purge and vent actions specified in EOP-3A to control PC pressure below the PC high pressure scram setpoint does not meet this threshold because such action is only permitted if off-site radioactivity release rates will remain below ODAM limits (ref. 1, 2, 3).

Intentional venting of PC for PC pressure (PCPL) or combustible gas control to the secondary containment and/or the environment per EOP-3A, SAG 1 or SAG 2 is a Loss of the PC barrier (ref. 1).

There is no PC Potential Loss threshold associated with PC Integrity or Bypass.

Reference(s):

1. EOP-3A, Primary Containment Control.
2. SAG 1, RPV, Containment and Radioactivity Release Control - Severe Accident Procedure 5.9SAMG Attachment 1.
3. SAG 2, RPV and PC Water Addition - Severe Accident Procedure 5.9SAMG Attachment 1.
4. EPFAQ 2015-005.
5. NEI 99-01, Primary Containment Isolation Failure PC Loss 3.B.

Barrier: Primary Containment

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

PC11
Any condition in the opinion of the Emergency Director that indicates loss of the PC Barrier

Definition(s):

None

Basis:

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

Reference(s):

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

Barrier: Primary Containment

Category: E. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

PC12

Any condition in the opinion of the Emergency Director that indicates potential loss of the PC Barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the PC 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.

Reference(s):

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

Category H – Hazards and Other Conditions Affecting Plant Safety

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

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

1. Security

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

2. Seismic Event

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

3. Natural or 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 plant 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.

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7. Emergency Director 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 the 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):

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 CNS or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILES, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should **not** be construed to include acts of civil disobedience or felonious acts that are **not** part of a concerted attack on CNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

OWNER CONTROLLED AREA (OCA) (Security) - The OCA for security purposes has been defined as the area immediately outside the vehicle barriers on the North and West side of the facility approximately 200 yards and out to the vehicle barriers on the south side of the facility.

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

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):

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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.

Basis:

This EAL 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 10CFR73.71 or 10CFR50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1 and HS1.

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

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

The first threshold references the Security Shift Supervision because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10CFR2.39 information.

The second threshold addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the Security Plan for CNS.

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 CNS procedure 5.5AIRCRAFT (ref. 2).

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

Escalation of the emergency classification level would be via EAL HA1.1.

Reference(s):

1. CNS Security and Safeguards Contingency Plan.
2. Procedure 5.5AIRCRAFT.
3. 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 (Security) or airborne attack threat within 30 minutes

EAL:

HA1.1 Alert

A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA (Security) 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):

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 CNS or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILES, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should **not** be construed to include acts of civil disobedience or felonious acts that are **not** part of a concerted attack on CNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

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.

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

OWNER CONTROLLED AREA (OCA) (Security) - The OCA for security purposes has been defined as the area immediately outside the vehicle barriers on the North and West side of the facility approximately 200 yards and out to the vehicle barriers on the south side of the facility.

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Basis:

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

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

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

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

This EAL 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 10CFR73.71 or 10CFR50.72.

The first threshold is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA (Security).

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 CNS procedure 5.5AIRCRAFT (ref. 2) .

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.

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In some cases, it may not be readily apparent if an aircraft impact within the OWNER CONTROLLED AREA (Security) 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 CNS Security and Safeguards Contingency Plan (ref. 1).

Escalation of the emergency classification level would be via EAL HS1.1.

Reference(s):

1. CNS Security and Safeguards Contingency Plan.
2. Procedure 5.5AIRCRAFT.
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):

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 CNS or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILES, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should **not** be construed to include acts of civil disobedience or felonious acts that are **not** part of a concerted attack on CNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

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.

OWNER CONTROLLED AREA (OCA) (Security) - The OCA for security purposes has been defined as the area immediately outside the vehicle barriers on the North and West side of the facility approximately 200 yards and out to the vehicle barriers on the south side of the facility.

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 (i.e., the security fence) and to which access is controlled.

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Basis:

This EAL addresses the occurrence of a HOSTILE ACTION within the PROTECTED AREA. This event will require rapid response and assistance due to the possibility for damage to plant equipment.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1).

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

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Site Area Emergency declaration will mobilize 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 EAL 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 10CFR73.71 or 10CFR50.72.

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 Security Plan for CNS (ref. 1).

Reference(s):

1. CNS Security and Safeguards Contingency Plan.
2. NEI 99-01 HS1.

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 > 0.1 g (OBE) as indicated by **EITHER**:

- Alarm B-3/A-1, EMERGENCY SEISMIC HIGH LEVEL
- MI-R-ACL1 or MI-CPU-NCC1 SEISMIC HIGH LEVEL red light(s) on Seismic Monitoring System panel

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE) (ref. 1, 2, 3). 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.

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CNS seismic instrumentation actuates at 0.01 g. The free field seismic sensor, located in the metal enclosure north of the Intake Structure, provides a start signal to the Seismic Monitoring System when ground motion > 0.01 g is sensed. On receipt of this start signal, the Seismic Monitor System indicates or initiates (ref. 1, 2):

- An Event is in-progress.
- An Event has been recorded.
- Annunciator B-3/B-1, SEISMIC EVENT, alarms.
- Recording of the seismic signals from sensors north of the Intake Structure, in the Reactor Building NW Quad 859', and on the Reactor Building 1001' level are recorded
- If seismic activity exceeds 0.1 g (OBE) the following indications are received in the control room (ref. 2):
    - Alarm B-3/A-1, EMERGENCY SEISMIC HIGH LEVEL
    - MI-R-ACL1 SEISMIC HIGH LEVEL red light on the Seismic Monitoring System panel
    - MI-CPU-NCC1 SEISMIC HIGH LEVEL red light on the Seismic Monitoring System panel

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 0.10g). To avoid inappropriate emergency classification resulting from spurious actuation of the seismic instrumentation or felt motion not attributable to seismic activity, the Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the U.S. Geological Survey (USGS), check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

However, the USGS, National Earthquake Information Center (NEIC)) can confirm that an earthquake has occurred in the area of the plant. The NEIC can be contacted by calling **(303) 273-8500**. Select **option #1** and inform the analyst you wish to confirm recent seismic activity in the vicinity of CNS. If requested, provide the analyst with the following CNS coordinates: **40° 21' north latitude, 95° 38' west longitude** (ref. 4).

Alternatively, near real-time seismic activity can be accessed via the NEIC website:

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

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Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

Reference(s):

1. Emergency Procedure 5.1Quake, Earthquake.
2. Alarm Procedure 2.3_B-3, Panel B - Annunciator B-3/A-1.
3. USAR Section II-5.2.4 and Table II-5-1.
4. USAR Section II-2-1.
5. 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 (i.e., the security fence) and to which access is controlled.

Basis:

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

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

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

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

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

Reference(s):

1. Emergency Procedure 5.1WEATHER, Operation During Weather Watches and Warnings.
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:

HU3.2 Unusual Event

Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component required by Technical Specifications 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:

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

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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 Categories A, F, S or C.

Refer to EAL CA6.1 or SA8.1 for internal FLOODING affecting more than one SAFETY SYSTEM train.

Reference(s):

1. Emergency Procedure 5.1FLOOD, Flood.
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 event external to the PROTECTED AREA involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

Basis:

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

This EAL addresses a hazardous materials event originating at a location outside the PROTECTED AREA 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 Categories A, F, S or C.

Reference(s):

1. Administrative Procedure 0.36.6, Monitoring for Industrial Gases.
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:

HU3.4 Unusual Event

A hazardous event that results in 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):

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.

Basis:

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

This EAL addresses a hazardous event that causes an 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 roads.

This EAL is not intended to 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 Categories A, F, S or C.

Reference(s):

1. NRC Event No. 53934.
2. 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Table H-1 Fire Areas
• PC (Drywell and Torus) • Reactor Building • Control Building • Service Water Pump Room • Diesel Generator Building • Cable Expansion Room

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.

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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.

Basis:

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

The intent of the HU4.1 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 the initial fire alarms, indication, or report. The 15-minute requirement begins with a credible notification that a FIRE is occurring, or receipt of multiple VALID fire detection system alarms or field validation of a single fire alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

Table H-1 Fire Areas are those areas that contain equipment necessary for safe operation and shutdown of the plant (ref. 1).

Reference(s):

1. USAR Section XII-2.1.2.1, Principal Class I Structures Required for Safe Shutdown.
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 (proved or disproved) within 30 min. of alarm receipt (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Table H-1 Fire Areas
• PC (Drywell and Torus) • Reactor Building • Control Building • Service Water Pump Room • Diesel Generator Building • Cable Expansion Room

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.

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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.

Basis:

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

HU4.2 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.

Note: While CNS is an NFPA-805 plant, the following excerpt from 10CFR50 Appendix R provides the bases for the acceptability of the use of a 30-minute validation time for a single fire zone alarm:

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Basis-Related Requirements from Appendix R

Appendix R to 10CFR50, 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 10CFR50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in HU4.2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

The 30-minute requirement begins upon receipt of a single VALID fire detection system alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field. Actual field reports must be made within the 30-minute time limit or a classification must be made. If a fire is verified to be occurring by field report, classification shall be made based on EAL HU4.1, with the 15-minute requirement beginning with the verification of the fire by field report.

Table H-1 Fire Areas are those areas that contain equipment necessary for safe operation and shutdown of the plant (ref. 1).

Reference(s):

1. USAR Section XII-2.1.2.1, Principal Class I Structures Required for Safe Shutdown.
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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is 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 (i.e., the security fence) and to which access is controlled.

Basis:

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

In addition to a FIRE addressed by EAL HU4.1 or HU4.2, a FIRE within the plant PROTECTED AREA but outside a Table H-1 Fire 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 EAL CA6.1 or SA8.1.

Reference(s):

1. NEI 99-01 HU4.

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.4 Unusual Event

A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

Basis:

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

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

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

Reference(s):

1. NEI 99-01 HU4.

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gas

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 room or area

AND

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

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then **no** emergency classification is warranted.

Table H-2 Safe Operation & Shutdown Rooms/Areas

Room/Area	Mode Applicability
Turbine Building 903′ Controlled Corridor	1
Reactor Building 958′	3
Reactor Building 931′ RHR Hx Room 1B	
Reactor Building 931′ RHR Hx Room 1A	
Reactor Building 931′ RWCU Hx Room	
Reactor Building 931′ General Area	
Reactor Building 903′ General Area	
Reactor Building 903′ RHR Hx Room 1A	
Reactor Building 903′ RHR Hx Room 1B	
Reactor Building 903′ Angle Valve Room	
Reactor Building 890′ Torus area	
Reactor Building 881′ NW Quad	
Control Building 882′ ECST room	
Reactor Building 881′ Torus Area	4

(continued on next page)

Mode Applicability:

1 – Power Operation, 3 – Hot Shutdown, 4 – Cold Shutdown

Definition(s):

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

Basis:

This EAL 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).

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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 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 generate smoke and 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 Category A, C or F ICs.

Reference(s):

1. Attachment 2 Safe Operation & Shutdown Rooms/Areas Tables A-2 & H-2 Bases.
2. NEI 99-01 HA5.

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

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

EAL:

HA6.1 Alert

An event results in plant control being transferred from the Control Room to the alternate shutdown locations

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL 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.

Transfer of plant control begins when the last licensed operator leaves the Control Room.

Escalation of the emergency classification level would be via EAL HS6.1.

Reference(s):

1. Emergency Procedure 5.1ASD, Alternate Shutdown.
2. Emergency Procedure 5.4FIRE-S/D, Fire Induced Shutdown From Outside Control Room.
3. 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 results in plant control being transferred from the Control Room to the alternate shutdown locations

AND

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

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Mode Applicability:

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

Definition(s):

None

Basis:

This EAL 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.

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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).

Transfer of plant control and the time period to establish control begins when the last licensed operator leaves the Control Room.

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

Reference(s):

1. Emergency Procedure 5.1ASD, Alternate Shutdown.
2. Emergency Procedure 5.4FIRE-S/D, Fire Induced Shutdown From Outside Control Room.
3. NRC EPFAQ 2015-014.
4. NEI 99-01 HS6.

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

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

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.

Reference(s):

1. NEI 99-01 HU7.

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director 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):

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 CNS or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILES, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should **not** be construed to include acts of civil disobedience or felonious acts that are **not** part of a concerted attack on CNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

OWNER CONTROLLED AREA (OCA) (Security) - The OCA for security purposes has been defined as the area immediately outside the vehicle barriers on the North and West side of the facility approximately 200 yards and out to the vehicle barriers on the south side of the facility.

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 (i.e., the security fence) and to which access is controlled.

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Basis:

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

Reference(s):

1. NEI 99-01 HA7.

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist 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):

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 CNS or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILES, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should **not** be construed to include acts of civil disobedience or felonious acts that are **not** part of a concerted attack on CNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

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

OWNER CONTROLLED AREA (OCA) (Security) - The OCA for security purposes has been defined as the area immediately outside the vehicle barriers on the North and West side of the facility approximately 200 yards and out to the vehicle barriers on the south side of the facility.

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SITE BOUNDARY - That boundary defined by a 1-mile radius around the plant.

Basis:

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

Reference(s):

1. NEI 99-01 HS7.

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist that 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):

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 CNS or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILES, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should **not** be construed to include acts of civil disobedience or felonious acts that are **not** part of a concerted attack on CNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

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.

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 (i.e., the security fence) and to which access is controlled.

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OWNER CONTROLLED AREA (OCA) (Security) - The OCA for security purposes has been defined as the area immediately outside the vehicle barriers on the North and West side of the facility approximately 200 yards and out to the vehicle barriers on the south side of the facility.

Basis:

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

Reference(s):

1. NEI 99-01 HG7.

Category S – System Malfunction

EAL Group: Hot Conditions; 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 Critical AC Power

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

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 125V DC vital buses.

3. Loss of Control Room Indications

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

4. RCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant rise from these base-line levels (> 1% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

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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 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 scram failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and containment integrity.

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 ESF AC Power

Initiating Condition: Loss of **all** offsite AC power capability to critical buses for 15 minutes or longer

EAL:

SU1.1 Unusual Event

Loss of **all** offsite AC power capability, Table S-1, to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Table S-1 AC Power Sources
Offsite • Startup Station Service Transformer • Emergency Station Service Transformer • Back-feed 345 kV line via Main Power Transformer to Normal Station Service Transformer (if already established) **Onsite** • DG-1 • DG-2 • Main Generator

Mode Applicability:

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

Definition(s):

None

Basis:

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

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

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

The 4160V Critical Buses 1F (Div 1) and 1G (Div 2) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following off-site sources of power (ref. 1, 2, 3, 4):

- Startup Transformer - The Startup Transformer provides a source of off-site AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of off-site AC power to the station whenever the main generator is off-line. The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the station Normal Transformer fails or that the main generator trips off-line.
- Emergency Transformer - The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by either the 69 kV transmission line from OPPD or the Cooper 161 kV transformer T-6. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.
- Back-feeding power from the 345 kV line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on-line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. The time required to establish the back-feed is likely longer than the 15-minute interval. If off-normal plant conditions have already established the back-feed its power to the safety-related buses may be considered an off-site power source.

On-site power sources, under hot plant conditions, are the emergency diesel generators (DG-1 and DG-2) and the Main Generator (ref. 2, 5).

Escalation of the emergency classification level would be via EAL SA1.1.

(continued on next page)

Reference(s):

1. System Operating Procedure 2.2.15, Startup Transformer.
2. System Operating Procedure 2.2.16, Normal Station Service Transformer.
3. System Operating Procedure 2.2.17, Emergency Station Service Transformer.
4. System Operating Procedure 2.2.18, 4160V Auxiliary Power Distribution System.
5. System Operating Procedure 2.2.20, Standby AC Power System (Diesel Generator).
6. NEI 99-01 SU1.

Category: S – System Malfunction

Subcategory: 1 – Loss of ESF AC Power

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

EAL:

SA1.1 Alert

AC power capability, Table S-1, to critical 4160V buses 1F and 1G reduced to a single power source for ≥ 15 min. (Note 1)

AND

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Table S-1 AC Power Sources
Offsite • Startup Station Service Transformer • Emergency Station Service Transformer • Back-feed 345 kV line via Main Power Transformer to Normal Station Service Transformer (if already established) **Onsite** • DG-1 • DG-2 • Main Generator

Mode Applicability:

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

(continued on next page)

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:

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

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

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

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

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The 4160V Critical Buses 1F (Div 1) and 1G (Div 2) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following off-site sources of power (ref. 1, 2, 3, 4):

- Startup Transformer - The Startup Transformer provides a source of off-site AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of off-site AC power to the station whenever the main generator is off-line. The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the station Normal Transformer fails or that the main generator trips off-line.
- Emergency Transformer - The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by either the 69 kV transmission line from OPPD or the Cooper 161 kV transformer T-6. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.
- Back-feeding power from the 345 kV line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on-line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. The time required to establish the back-feed is likely longer than the 15-minute interval. If off-normal plant conditions have already established the back-feed its power to the safety-related buses may be considered an off-site power source.

On-site power sources, under hot plant conditions, are the emergency diesel generators (DG-1 and DG-2) and the Main Generator (ref. 2, 5).

Escalation of the emergency classification level would be via EAL SS1.1.

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

(continued on next page)

Reference(s):

1. System Operating Procedure 2.2.15, Startup Transformer.
2. System Operating Procedure 2.2.16, Normal Station Service Transformer.
3. System Operating Procedure 2.2.17, Emergency Station Service Transformer.
4. System Operating Procedure 2.2.18, 4160V Auxiliary Power Distribution System.
5. System Operating Procedure 2.2.20, Standby AC Power System (Diesel Generator).
6. NEI 99-01 SA1.

Category: S – System Malfunction

Subcategory: 1 – Loss of ESF AC Power

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

EAL:

SS1.1 Site Area Emergency

Loss of **all** offsite and **all** onsite AC power to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

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:

This EAL 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.

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

The 4160V Critical Buses 1F (Div 1) and 1G (Div 2) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following off-site sources of power (ref. 1, 2, 3, 4):

- Startup Transformer - The Startup Transformer provides a source of off-site AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of off-site AC power to the station whenever the main generator is off-line. The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the station Normal Transformer fails or that the main generator trips off-line.
- Emergency Transformer - The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by either the 69 kV transmission line from OPPD or the Cooper 161 kV transformer T-6. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.
- Back-feeding power from the 345 kV line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on-line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. The time required to establish the back-feed is likely longer than the 15-minute interval. If off-normal plant conditions have already established the back-feed its power to the safety-related buses may be considered an off-site power source.

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On-site power sources, under hot plant conditions, are the emergency diesel generators (DG-1 and DG-2) and the Main Generator (ref. 2, 5).

Also under this EAL, provided the 15-minute interval can be met, credit can also be taken for non-safety-related power sources provided that operation of those sources is controlled in accordance with abnormal or emergency operating procedures (e.g., Supplemental Diesel Generator [SDG]) (ref. 6), or beyond design basis accident response guidelines (e.g., FLEX support guidelines) (ref. 7, 8). Such power sources should generally meet the "Alternate ac source" definition provided in 10CFR50.2.

Escalation of the emergency classification level would be via IC AG1, FG1 or SG1.

This EAL is the hot condition equivalent of the cold condition EAL CA2.1.

Reference(s):

1. System Operating Procedure 2.2.15, Startup Transformer.
2. System Operating Procedure 2.2.16, Normal Station Service Transformer.
3. System Operating Procedure 2.2.17, Emergency Station Service Transformer.
4. System Operating Procedure 2.2.18, 4160V Auxiliary Power Distribution System.
5. System Operating Procedure 2.2.20, Standby AC Power System (Diesel Generator).
6. EPFAQ 2015-015.
7. FLEX Support Guideline 5.10FLEX.07 4160V "F" Bus Tie-In with Off-Site Generator.
8. FLEX Support Guideline 5.10FLEX.08 4160V "G" Bus Tie-In with Off-Site Generator.
9. NEI 99-01 SS1.

Category: S –System Malfunction

Subcategory: 1 – Loss of ESF AC Power

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

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power to critical 4160V buses 1F and 1G

AND EITHER:

- RPV water level **cannot** be restored and maintained > -183 in.
- Long-term RCS heat removal capability is **not** likely to be established and maintained per procedure 5.3SBO Station Blackout

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:

The EAL threshold is based on either of the following conditions due to a prolonged loss of all AC power to the critical busses:

- Inability to restore and maintain RPV water level above the Minimum Steam Cooling Reactor Water Level (ref. 9)
- The inability to establish and maintain long-term RCS heat removal capability per emergency procedure 5.3SBO (ref. 10).

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For extended loss of emergency bus AC power events that do not result in a breach of the RCS barrier, this 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.

The EAL will 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.

Indication of continuing core cooling degradation is manifested by the inability to restore and maintain RPV water level above the Minimum Steam Cooling Reactor Water Level (-183 in.) (ref. 9). Core submergence is the most desirable means of core cooling, however when RPV level is below TAF, the uncovered portion of the core can be cooled by less reliable means (i.e., steam cooling or spray cooling).

The term, "cannot be restored and maintained >," means the value of RPV water level is not able to be brought above the specified limit (MSCRWL). 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 beyond the limit**; the threshold must be considered reached as soon as it is apparent that the MSCRWL cannot be attained.

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

The EAL requires 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.

This EAL may 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.

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The 4160V Critical Buses 1F (Div 1) and 1G (Div 2) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following off-site sources of power (ref. 1, 2, 3, 4):

- Startup Transformer - The Startup Transformer provides a source of off-site AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of off-site AC power to the station whenever the main generator is off-line. The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the station Normal Transformer fails or that the main generator trips off-line.
- Emergency Transformer - The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by either the 69 kV transmission line from OPPD or the Cooper 161 kV transformer T-6. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.
- Back-feeding power from the 345 kV line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on-line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. The time required to establish the back-feed is likely longer than the 15-minute interval. If off-normal plant conditions have already established the back-feed its power to the safety-related buses may be considered an off-site power source.

On-site power sources, under hot plant conditions, are the emergency diesel generators (DG-1 and DG-2) and the Main Generator (ref. 2, 5).

Also under this EAL, credit can also be taken for non-safety-related power sources provided that operation of those sources is controlled in accordance with abnormal or emergency operating procedures (e.g., Supplemental Diesel Generator [SDG]) (ref. 6), or beyond design basis accident response guidelines (e.g., FLEX support guidelines) (ref. 7, 8). Such power sources should generally meet the "Alternate ac source" definition provided in 10CFR50.2.

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Reference(s):

1. System Operating Procedure 2.2.15, Startup Transformer.
2. System Operating Procedure 2.2.16, Normal Station Service Transformer.
3. System Operating Procedure 2.2.17, Emergency Station Service Transformer.
4. System Operating Procedure 2.2.18, 4160V Auxiliary Power Distribution System.
5. System Operating Procedure 2.2.20, Standby AC Power System (Diesel Generator).
6. EPFAQ 2015-015.
7. FLEX Support Guideline 5.10FLEX.07 4160V "F" Bus Tie-In with Off-Site Generator.
8. FLEX Support Guideline 5.10FLEX.08 4160V "G" Bus Tie-In with Off-Site Generator.
9. EOP-1A, RPV Control.
10. Emergency Procedure 5.3SBO, Station Blackout.
11. NEI 99-01 SG1.

Category: S – System Malfunction

Subcategory: 1 – Loss of ESF AC Power

Initiating Condition: Loss of **all** critical 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 to critical 4160V buses 1F and 1G for ≥ 15 min. (Note 1)

AND

Indicated voltage is < 105 VDC on vital 125 VDC bus 1A **AND** 1B for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

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.

(continued on next page)

Basis:

This IC addresses a concurrent and prolonged loss of both critical AC and Vital DC power. A loss of all critical 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 critical AC and Vital DC power will lead to multiple challenges to fission product barriers.

The 4160V Critical Buses 1F (Div 1) and 1G (Div 2) are the plant essential, safety-related emergency buses. Each can be energized manually and separately by any of the following off-site sources of power (ref. 1, 2, 3, 4):

- Startup Transformer - The Startup Transformer provides a source of off-site AC power to the entire Auxiliary Power Distribution System adequate for the startup operation or shutdown operation of the station. The Startup Transformer is the preferred source of off-site AC power to the station whenever the main generator is off-line. The Startup Transformer is energized from the 161 kV Switchyard. The transformer is normally left energized at all times to provide for quick automatic transfer of the 4160V auxiliaries to the Startup Transformer in the event that the station Normal Transformer fails or that the main generator trips off-line.
- Emergency Transformer - The Emergency Transformer is the primary off-site AC power source to essential station loads. During normal station operation, the Emergency Transformer is energized by either the 69 kV transmission line from OPPD or the Cooper 161 kV transformer T-6. As such, it supplies 4160V Switchgear 1F and/or 1G in the event that the Normal Transformer and Startup Transformer are not available for service. Use of the Emergency Transformer also allows portions of the 345 kV System to be removed from service for inspection, testing, and maintenance.
- Back-feeding power from the 345 kV line through the Main Power Transformer to the Normal Transformer. The Normal Transformer is the normal source of AC power to the station when the Main Generator is on-line above 20% (160 MWe) electrical power. The transformer is energized during Main Generator operation through the Isolated Phase Buses that feed the Main Power Transformers. The time required to establish the back-feed is likely longer than the 15-minute interval. If off-normal plant conditions have already established the back-feed its power to the safety-related buses may be considered an off-site power source.

On-site power sources, under hot plant conditions, are the emergency diesel generators (DG-1 and DG-2) and the Main Generator (ref. 2, 5).

(continued on next page)

Also under this EAL, provided the 15-minute interval can be met, credit can also be taken for non-safety-related power sources provided that operation of those sources is controlled in accordance with abnormal or emergency operating procedures (e.g., Supplemental Diesel Generator [SDG]) (ref. 6), or beyond design basis accident response guidelines (e.g., FLEX support guidelines) (ref. 7, 8). Such power sources should generally meet the "Alternate ac source" definition provided in 10CFR50.2.

105 VDC is the minimum design bus voltage (ref. 9).

The 125 VDC System supplies DC power to conventional station emergency equipment and selected Safeguard System loads. 125 VDC Distribution Panels supply control and instrument power for annunciators, control logic power and protective relaying.

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

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

Battery chargers receive their power from 460V critical motor control centers. Each 125 VDC bus receives power from either a 125 VDC battery or a 125 VDC battery charger. The 250 VDC System supplies DC power to conventional station emergency equipment and selected Safeguard System loads. Although 250 VDC Buses 1A and 1B provide vital DC emergency power, 250 VDC Safety System loads (such as motor operated valves) also require 125 VDC control power. Loss of 125 VDC buses alone, therefore, would render most Safeguard System loads inoperable (ref. 10, 11, 12).

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.

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Reference(s):

1. System Operating Procedure 2.2.15, Startup Transformer.
2. System Operating Procedure 2.2.16, Normal Station Service Transformer.
3. System Operating Procedure 2.2.17, Emergency Station Service Transformer.
4. System Operating Procedure 2.2.18, 4160V Auxiliary Power Distribution System.
5. System Operating Procedure 2.2.20, Standby AC Power System (Diesel Generator).
6. EPFAQ 2015-015.
7. FLEX Support Guideline 5.10FLEX.07 4160V "F" Bus Tie-In with Off-Site Generator.
8. FLEX Support Guideline 5.10FLEX.08 4160V "G" Bus Tie-In with Off-Site Generator.
9. Technical Specifications B 3.8.4.
10. USAR Section VIII-6.2.
11. USAR Section VIII-6.3.
12. Emergency Procedure 5.3DC125, Loss of 125 VDC.
13. 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
Indicated voltage is < 105 VDC on vital 125 VDC bus 1A **AND** 1B for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

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:

This EAL 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.

105 VDC is the minimum design bus voltage (ref. 1).

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The 125 VDC System supplies DC power to conventional station emergency equipment and selected Safeguard System loads. 125 VDC Distribution Panels supply control and instrument power for annunciators, control logic power and protective relaying.

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

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

Battery chargers receive their power from 460V critical motor control centers. Each 125 VDC bus receives power from either a 125 VDC battery or a 125 VDC battery charger. The 250 VDC System supplies DC power to conventional station emergency equipment and selected Safeguard System loads. Although 250 VDC Buses 1A and 1B provide vital DC emergency power, 250 VDC Safety System loads (such as motor operated valves) also require 125 VDC control power. Loss of 125 VDC buses alone, therefore, would render most Safeguard System loads inoperable (ref. 2, 3, 4).

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

Escalation of the emergency classification level would be via IC AG1, FG1 or SG1.

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

Reference(s):

1. Technical Specifications B 3.8.4.
2. USAR Section VIII-6.2.
3. USAR Section VIII-6.3.
4. Emergency Procedure 5.3DC125, Loss of 125 VDC.
5. 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

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

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.

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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:

This EAL 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 10CFR50.72 (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 water 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 EAL SA3.1.

Reference(s):

1. USAR Section VII-1 Control and Instrumentation.
2. 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 significant transient is in progress, Table S-3

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

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

Table S-3 Significant Transients
• Reactor scram • Runback > 25% thermal power • Electrical load rejection > 25% full electrical load • ECCS injection • Thermal power oscillations > 10%

(continued on next page)

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.

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:

This EAL 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 10CFR50.72 (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.

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This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, RPV water 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 FS1 or AS1

Reference(s):

1. USAR Section VII-1 Control and Instrumentation.
2. NEI 99-01 SA2.

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

SU4.1 Unusual Event
SJAE monitor RMP-RM-150A(B) > 1.58E+3 mR/h (Hi-Hi setpoint)

Mode Applicability:

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

Definition(s):

None

Basis:

Steam Jet Air Ejectors (SJAEs) remove all non-condensable gases from the condensers including air in-leakage and disassociated products originating in the reactor and exhausts them to the off-gas holdup volume. A rise in off-gas activity could therefore indicate damage to the fuel cladding, a potential degradation in the level of safety of the plant, and a potential precursor of more serious problems. The SJAE monitor Hi-Hi radiation setpoint is set at 50% of the instantaneous release limit and represents approximately 0.1% (Technical Specification limit) fuel cladding damage (ref. 5). The SJAE monitor Hi-Hi radiation setpoint (1.58E+3 mR/hr) has been selected because it is operationally significant and is readily recognizable by the Control Room Operating Staff (ref. 1, 2, 3, 4). The Off-Gas System isolates after a 15-minute time delay (ref. 1, 2).

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 IC FA1 or the Category A ICs.

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Reference(s):

1. Alarm Procedure 2.3_9-4-1, Panel 9-4 - Annunciator 9-4-1, C-4, OFFGAS TIMER INITIATED.
2. Alarm Procedure 2.3_9-4-1, Panel 9-4 - Annunciator 9-4-1, C-5, OFFGAS HIGH RAD.
3. Abnormal Procedure 2.4OG, Off-Gas Abnormal.
4. Emergency Procedure 5.2FUEL, Fuel Failure.
5. NEDC 02-004, Estimation of the Steam Jet Air Ejector Radiation Monitor, RMP-RM-150A(B), Readings Following a 1% Fuel Clad Release (Degraded Core) in the Reactor Coolant System.
6. NEI 99-01 SU3.

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

SU4.2 Unusual Event
Coolant activity ≥ 4.0 μCi/gm dose equivalent I-131

Mode Applicability:

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

Definition(s):

None

Basis:

This EAL addresses a reactor coolant activity value that exceeds the allowable transient limit specified in Technical Specifications LCO 3.4.6. 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 IC FA1 or the Category A ICs.

Reference(s):

1. Technical Specification LCO 3.4.6.
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. (Note 1)

OR

RCS identified leakage > 25 gpm for ≥ 15 min. (Note 1)

OR

Leakage from the RCS to a location outside PC > 25 gpm for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Mode Applicability:

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

Definition(s):

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

Basis:

Failure to isolate the leak within 15 minutes, or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

This EAL 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.

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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) (ref. 1, 2). 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.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. An emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated). 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 Category A or F.

Reference(s):

1. Technical Specification Definitions Section 1.1.
2. Technical Specification LCO 3.4.4, RCS Operational Leakage.
3. 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 as indicated by reactor power > 3% after **any** RPS setpoint is exceeded

AND

A subsequent automatic scram or manual scram action taken at Panel 9-5 (Mode Switch, Manual PBs, ARI) is successful in shutting down the reactor as indicated by reactor power ≤ 3% (APRM downscale) (Note 8)

Note 8: A manual scram action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation, 2 - Startup

Definition(s):

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:

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 console (Panel 9-5) 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.

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function. A reactor scram is automatically initiated by the Reactor Protection System (RPS) when certain continuously monitored parameters exceed predetermined setpoints (ref. 1).

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A successful scram has occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power to or below the APRM downscale setpoint of 3% (ref. 2).

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console, Panel 9-5, (i.e., Mode Switch, manual scram pushbuttons, or ARI initiation). Reactor shutdown achieved by use of alternate control rod insertion methods does not constitute a successful manual scram (ref. 3).

Following any automatic RPS scram signal, operating procedures (e.g., EOP-1A/6A) prescribe insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved. Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Unusual Event.

Taking the Mode Switch to Shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

For the purposes of this EAL, a successful <u>automatic</u> initiation of ARI that reduces reactor power ≤ 3% is <u>not</u> considered a successful automatic scram. If automatic initiation of ARI has occurred and caused reactor shutdown, the automatic RPS scram must have failed. ARI is a backup means of inserting control rods in the unlikely event that an automatic RPS scram signal exists but the reactor continues to generate significant power. However, a successful automatic or manual initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

In the event that the operator identifies a reactor scram is IMMINENT and initiates a successful manual reactor scram before the automatic scram setpoint is reached, no declaration is required. The successful manual scram of the reactor before it reaches its automatic scram setpoint or reactor scram signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. If manual reactor scram actions fail to reduce reactor power to ≤ 3%, the event escalates to the Alert under EAL SA6.1.

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If by procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

Following the failure of 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 or manually initiating ARI are considered to be a manual scram actions.

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 EAL SA6.1. 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.

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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 generated as a result of plant work causes a plant transient that results in a condition that should have included an automatic reactor scram and the RPS fails to automatically shutdown the reactor, then this IC and associated EALs are applicable, and should be evaluated.
- If the signal generated as a result of plant work 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 associated EALs are not applicable and no classification is warranted.

Reference(s):

1. General Operating Procedure 2.1.5, Reactor Scram.
2. AMP-TBD00 Technical Basis PSTG/SATG.
3. Emergency Procedure 5.8.3, Alternate Rod Insertion Methods.
4. NEI 99-01 SU5.

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor

EAL:

SU6.2 Unusual Event

A manual scram did **not** shut down the reactor as indicated by reactor power > 3% after **any** manual scram action was initiated

AND

A subsequent automatic scram or manual scram action taken at Panel 9-5 (Mode Switch, Manual PBs, ARI) is successful in shutting down the reactor as indicated by reactor power ≤ 3% (APRM downscale) (Note 8)

Note 8: A manual scram action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation, 2 - Startup

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:

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.

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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%) (ref. 1).

A successful scram has occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power to or below the APRM downscale setpoint of 3% (ref. 2).

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., Mode Switch, manual scram pushbuttons, or ARI initiation). Reactor shutdown achieved by use of alternate control rod insertion methods does not constitute a successful manual scram (ref. 3).

Taking the Mode Switch to Shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

Successful automatic or manual initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

If both subsequent automatic and subsequent manual reactor scram actions in the Control Room fail to reduce reactor power below the power associated with the SAFETY SYSTEM design (≤ 3%) following a failure of an initial manual scram, the event escalates to an Alert under EAL SA6.1.

Following the failure of 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.

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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 EAL SA6.1. 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 generated as a result of plant work causes a plant transient that results in a condition that should have included an automatic reactor scram and the RPS fails to automatically shutdown the reactor, then this IC and associated EALs are applicable, and should be evaluated.
- If the signal generated as a result of plant work 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 associated EALs are not applicable and no classification is warranted.

Reference(s):

1. General Operating Procedure 2.1.5, Reactor Scram.
2. AMP-TBD00 Technical Basis PSTG/SATG.
3. Emergency Procedure 5.8.3, Alternate Rod Insertion Methods.
4. NEI 99-01 SU5.

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor 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 as indicated by reactor power > 3%

AND

Manual scram actions taken at Panel 9-5 (Mode Switch, Manual PBs, ARI) are **not** successful in shutting down the reactor as indicated by reactor power > 3% (Note 8)

Note 8: A manual scram action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation, 2 - Startup

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.

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Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control console, Panel 9-5, 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 Panel 9-5 since this event entails a significant failure of the RPS.

This EAL addresses any automatic or manual reactor scram signal that fails to shut down the reactor followed by subsequent manual scram actions that fail to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the SAFETY SYSTEMS were designed (> 3%) (ref. 1, 2).

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console, Panel 9-5, (i.e., Mode Switch, manual scram pushbuttons, or ARI initiation). Reactor shutdown achieved by use of alternate control rod insertion methods does not constitute a successful manual scram (ref. 3).

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.

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 consoles (e.g., locally venting HCUs). 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.

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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 Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

Reference(s):

1. General Operating Procedure 2.1.5, Reactor Scram.
2. AMP-TBD00 Technical Basis PSTG/SATG.
3. Emergency Procedure 5.8.3, Alternate Rod Insertion Methods.
4. 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 as indicated by reactor power > 3%

AND

All actions to shut down the reactor are **not** successful as indicated by reactor power > 3%

AND EITHER:

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

Heat Capacity Temperature Limit (HCTL) exceeded (EOP/SAG Graph 7)

Mode Applicability:

1 - Power Operation, 2 - Startup

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.

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Basis:

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

This EAL addresses the following:

- Any automatic reactor scram signal followed by subsequent manual scram actions 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 (EAL SA6.1), and
- Indications that either core cooling is extremely challenged or PC heat removal is extremely challenged (ref. 4, 5).

Reactor shutdown achieved by use of any control rod insertion methods (ref. 3) or boron injection are also credited as a successful manual scram provided reactor power can be reduced below the APRM downscale trip setpoint before indications of an extreme challenge to either core cooling or PC heat removal exist (ref. 2).

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 potential threat to the Fuel Clad, RCS and PC barriers.

Indication that core cooling is extremely challenged is manifested by inability to restore and maintain RPV water level above the Minimum Steam Cooling RPV Water Level (MSCRWL) (ref. 1). The MSCRWL (-183 in.) is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500°F. This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core steam flow, could be a precursor of a core melt sequence (ref. 2).

The Heat Capacity Temperature Limit (HCTL, EOP/SAG Graph 7) is the highest torus water temperature from which Emergency RPV Depressurization will not raise torus temperature above the maximum design torus temperature.

The HCTL 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 (ref. 5). This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature.

(continued on next page)

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

Escalation of the emergency classification level would be via IC AG1 or FG1.

Reference(s):

1. General Operating Procedure 2.1.5, Reactor Scram.
2. AMP-TBD00 Technical Basis PSTG/SATG.
3. Emergency Procedure 5.8.3, Alternate Rod Insertion Methods.
4. EOP-1A RPV Control.
5. EOP-3A, Primary Containment Control.
6. 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** communication methods, Table S-4, for **any** of the following:
- Onsite
- State/Local
- NRC

Table S-4 Communication Methods

System	Onsite	State/Local	NRC
Station Intercom System (Gaitronics)	X		
Site UHF Radio Consoles	X		
Alternate Intercom	X		
CNS On-Site Cell Phone System	X	X	X
Telephone System (PBX)	X	X	X
Local Telephones (C. O. Lines)		X	X
Satellite Telephones		X	X
CNS State Notification Telephones		X	
Federal Telecommunications System (FTS 2001)			X

(continued on next page)

Mode Applicability:

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

Definition(s):

None

Basis:

This EAL addresses a significant loss of on-site or offsite communications capabilities (ref. 1, 2). While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to State and local agencies and the NRC.

This EAL 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 bullet addresses a total loss of the communications methods used in support of routine plant operations.

The second bullet addresses a total loss of the communications methods used to notify all State and local agencies of an emergency declaration. The State and local agencies referred to here are the Nebraska State Patrol, Missouri State Patrol, Atchison County Sheriff's Department and the Nemaha and Richardson County Sheriff's Departments.

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

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

Reference(s):

1. EPIP 5.7COMMUN, Communications, Emergency Response Facility Communication Equipment attachment.
2. Emergency Plan for Cooper Nuclear Station, Section 7.3 Communications Systems and Notification.
3. NEI 99-01 SU6.

Category: S – System Malfunction

Subcategory: 8 – Hazardous Event Affecting Safety Systems

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

EAL:

SA8.1 Alert

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

AND

Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode

AND EITHER:

- Event damage has caused indications of degraded performance to the redundant train of the SAFETY SYSTEM needed for the current operating mode
- Event damage has resulted in VISIBLE DAMAGE to the redundant train of the SAFETY SYSTEM needed for the current operating mode

(Notes 9, 10)

Note 9: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then emergency classification is **not** warranted.

Note 10: If the hazardous event **only** resulted in VISIBLE DAMAGE, with **no** indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is **not** warranted.

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

(continued on next page)

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 SAFETY SYSTEM train 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 SAFETY SYSTEM train.

(continued on next page)

Basis:

This EAL addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second (redundant) SAFETY SYSTEM train or VISIBLE DAMAGE to the redundant train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second (redundant) SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues.

Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance 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.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make a determination of VISIBLE DAMAGE 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. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via IC FS1 or AS1.

This EAL is the hot condition equivalent of the cold condition EAL CA6.1.

Reference(s):

1. EP FAQ 2016-002.
2. NEI 99-01 SA9.

ATTACHMENT 3 SAFE OPERATION AND SHUTDOWN ROOMS/AREAS TABLES A-2 AND H-2 BASES [INFORMATION USE]

3. BACKGROUND

3.1 NEI 99-01, Revision 6 ICs AA3 and HA5 prescribe declaration of an Alert based on impeded access to rooms or areas (due to either area radiation levels or hazardous gas concentrations) where equipment necessary for normal plant operations, cooldown or shutdown is located. These areas are intended to be plant operating mode dependent. Specifically, the Developers Notes for AA3 and HA5 states:

3.1.1 The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.

4. The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).

5. Further, as specified in IC HA5:

5.1 The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.

6. CNS TABLE A-2 AND H-2 BASES

6.1 A review of station operating procedures identified the following mode dependent in-plant actions and associated areas that are required for normal plant operation, cooldown or shutdown:

Location- Operational Area	Actions and Modes
Reactor Building 1001	No entry required
Reactor Building 976′ East	No entry required
Reactor Building 976′ West	No entry required
Reactor Building 958′	Required for venting RHR Subsystem B for placing in SDC and isolation of condensate makeup (Mode 3)
Reactor Building 931′ RHR Hx Room 1B	Required for venting RHR subsystem B for placing in SDC (Mode 3)
Reactor Building 931′ RHR Hx Room 1A	Required for Venting RHR subsystem A for placing in SDC (Mode 3)
Reactor Building 931′ RWCU Hx Room	Required to align RWCU Subcooling line to support continued operation of RWCU for RPV inventory reject flow path. (Mode 3)
Reactor Building 931′ General Area	Required to control cooling to RWCU NRHX for RPV inventory reject flow path. (Mode 3)
Reactor Building 903′ General Area	Required for opening RHR Subsystem A(B) minimum flow valve power supply breaker in support of placing in SDC. (Mode 3)
Reactor Building 903′ RHR Hx Room 1A	Required for venting RHR subsystem A, aligning for flushing / heat up of SDC, and Chemistry Sampling. (Mode 3)
Reactor Building 903′ RHR Hx Room 1B	Required for venting RHR Subsystem B. (Mode 3)
Reactor Building 903′ Angle Valve Room	Required for venting SDC. (Mode 3)
Reactor Building 890′ Torus area	Required for venting SDC. (Mode 3)
Reactor Building 881′ Torus Area	Required for aligning pneumatics to the RPV Vent isolation valves. (Mode 4)
Reactor Building 881′ NW Quad	Required for aligning condensate makeup to support flushing / heat up of SDC. (Mode 3)
Control Building 918′	No entry required
Control Building 903′	No entry required
Control Building 882′ SWBP Area	No entry required
Control Building 882′ ECST room	Required for operating disconnect for RHR SDC PCIV. (Mode 3)
Service Water Pump Room	No entry required
EDG #1 Room	No entry required
EDG #1 Day Tank Room	No entry required
EDG #2 Room	No entry required
EDG #2 Day Tank Room	No entry required
Turbine Building 882′	No entry required
Turbine Building 903′ Controlled Corridor	Entry Required for installation of bug sources on SJAE Rad Monitors (Mode 1)
Turbine Building 903′ and 909′ Heater Bay	No entry required
Turbine Building 932′	No entry required
Rad Waste Control Room	No entry required

6.2 Control Room ventilation systems have adequate engineered safety/design features in place to preclude a Control Room evacuation due to the release of a hazardous gas. Therefore, the Control Room is not included in this assessment or in Table H-2.

7. PLANT OPERATING PROCEDURES REVIEWED

7.1 GENERAL OPERATING PROCEDURES

- Procedure 2.1.4, Normal Shutdown.
- Procedure 2.1.4.1, Rapid Shutdown.
- Procedure 2.1.5, Reactor Scram.

7.2 SYSTEM OPERATING PROCEDURES

- Procedure 2.2.5, Condensate Filter Demineralizer System.
- Procedure 2.2.6, Condensate System.
- Procedure 2.2.8, Control Rod Drive Hydraulic System.
- Procedure 2.2.28, Feedwater System Startup and Shutdown.
- Procedure 2.2.28.1, Feedwater System Operation.
- Procedure 2.2.55, Main Condenser Gas Removal System.
- Procedure 2.2.56, Main Steam System.
- Procedure 2.2.58, AOG System
- Procedure 2.2.58.3, AOG System Operation with Recombiner A.
- Procedure 2.2.58.4, AOG System Operation with Recombiner B.
- Procedure 2.2.66, Reactor Water Cleanup.
- Procedure 2.2.68.1, Reactor Recirculation System Operations.
- Procedure 2.2.69.2, RHR System Shutdown Operations.
- Procedure 2.2.70, RHR Service Water Booster Pump System.
- Procedure 2.2.75, Steam Sealing System.
- Procedure 2.2.77, Main Turbine.
- Procedure 2.2.78, Turbine Cylinder Heating System.

8. TABLE A-2 & H-2 RESULTS

Table A-2 & H-2Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Turbine Building 903′ Controlled Corridor	1
Reactor Building 958′	3
Reactor Building 931′ RHR Hx Room 1B	
Reactor Building 931′ RHR Hx Room 1A	
Reactor Building 931′ RWCU Hx Room	
Reactor Building 931′ General Area	
Reactor Building 903′ General Area	
Reactor Building 903′ RHR Hx Room 1A	
Reactor Building 903′ RHR Hx Room 1B	
Reactor Building 903′ Angle Valve Room	
Reactor Building 890′ Torus area	
Reactor Building 881′ NW Quad	
Control Building 882′ ECST room	
Reactor Building 881′ Torus Area	4

ATTACHMENT 4 EAL CLASSIFICATION MATRIX [INFORMATION USE]

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

- EAL Identifier.
- Mode Applicability.
- EAL.

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

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

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

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

EAL Classification Matrix Revision Data:	
Procedure	EAL Classification Matrix Revision Number
EPIP 5.7.1, Attachment 4	Revision 22

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

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

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

EAL Classification Matrix Locations:

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

ATTACHMENT 5 EAL DEFINITIONS AND ACRONYMS [INFORMATION USE]

DEFINITIONS

Selected terms used in Initiating Condition, Emergency Action Level statements and EAL bases 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. (ref 3.1.1 except as noted)

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 Protective Action Guideline exposure levels.

Confinement Boundary

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the CNS ISFSI, Confinement Boundary is defined as the NUHOMS Dry Shielded Canister (DSC) (ref. 3.1.10).

Containment Closure

The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Secondary Containment Closure requirements are defined in AP 0.50.5 Outage Shutdown Safety (ref. 3.1.13).

Emergency Action Level (EAL)

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

Emergency Classification Level (ECL)

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

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

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

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.

Impede(d)

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

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 (IC)

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

Owner Controlled Area (OCA) (Security)

The OCA for security purposes has been defined as the area immediately outside the vehicle barriers on the North and West side of the facility approximately 200 yards and out to the vehicle barriers on the south side of the facility (ref. 3.1.12).

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 (i.e., the security fence) and to which access is controlled (ref. 3.1.11).

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

Reactor cavity, fuel transfer canal (cattle chute), and spent fuel pool, but not including the reactor vessel, comprise the refueling pathway (ref 3.1.13).

Restore

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

Safety System

A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

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

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

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

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

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 boundary defined by a 1-mile radius around the plant (ref. 3.1.14).

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 (UE)

Events are in progress or have occurred which indicate a potential degradation in 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 SAFETY SYSTEM train 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 SAFETY SYSTEM train.

Abbreviations/Acronyms

°F ..... Degrees Fahrenheit
AC ..... Alternating Current
ADS ..... Automatic Depressurization System
AOP ..... Abnormal Operating Procedure
APRM ..... Average Power Range Meter
ARI ..... Alternate Rod Insertion
ATWS ..... Anticipated Transient Without Scram
BWR ..... Boiling Water Reactor
BWROG ..... Boiling Water Reactor Owners Group
CAS ..... Central Alarm Station
CDE ..... Committed Dose Equivalent
CFR ..... Code of Federal Regulations
CNS ..... Cooper Nuclear Station
CR ..... Control Room
CS ..... Core Spray
DEF ..... Defueled
DC ..... Direct Current
DG ..... Diesel Generator
DSC ..... Dry Storage Canister
EAL ..... Emergency Action Level
ECCS ..... Emergency Core Cooling System
ECL ..... Emergency Classification Level
EOP ..... Emergency Operating Procedure
EPA ..... Environmental Protection Agency
EPG ..... Emergency Procedure Guideline
EPIP ..... Emergency Plan Implementing Procedure
ERP ..... Elevated Release Point
ESF ..... Engineered Safety Feature
FAA ..... Federal Aviation Administration
FAQ ..... Frequently Asked Question
FBI ..... Federal Bureau of Investigation
FC ..... Fuel Clad
Ft. ..... Feet
GE ..... General Emergency
HCTL ..... Heat Capacity Temperature Limit
HCU ..... Hydraulic Control Unit

HPCI ..... High Pressure Coolant Injection
H_2 ..... Hydrogen
Hr. ..... Hour
HSM ..... Horizontal Storage Module
IC ..... Initiating Condition
In. ..... Inch(es)
ISFSI ..... Independent Spent Fuel Storage Installation
LCO ..... Limiting Condition of Operation
LOCA ..... Loss of Coolant Accident
MCSF ..... Minimum Core Steam Flow
Min. ..... Minute
mR, mRem, mrem, mREM ..... milli-Roentgen Equivalent Man
MSCRWL ..... Minimum Steam Cooling RPV Water Level
MSIV ..... Main Steam Isolation Valve
MW ..... Megawatt
N/A ..... Not Applicable
NEI ..... Nuclear Energy Institute
NEIC ..... National Earthquake Information Center
NESP ..... National Environmental Studies Project
NORAD ..... North American Aerospace Defense Command
NRC ..... Nuclear Regulatory Commission
OBE ..... Operating Basis Earthquake
OCA ..... Owner Controlled Area
ODAM ..... Off-site Dose Assessment Manual
OPPD ..... Omaha Public Power District
ORO ..... Offsite Response Organization
PAG ..... Protective Action Guideline
PB ..... Pushbutton
PC ..... Primary Containment
PCPL ..... Primary Containment Pressure Limit
PMIS ..... Plant Monitoring & Information System
PRA/PSA ..... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PSIG ..... Pounds per Square Inch Gauge
R ..... Roentgen
RB ..... Reactor Building
RCIC ..... Reactor Core Isolation Cooling
RCS ..... Reactor Coolant System

SDG ..... Supplemental Diesel Generator
Rem, rem, REM ..... Roentgen Equivalent Man
RFPT ..... Reactor Feed Pump
RHR ..... Residual Heat Removal
RPS ..... Reactor Protection System
RPV ..... Reactor Pressure Vessel
RRMG ..... Recirculation Motor Generator
RWCU ..... Reactor Water Cleanup
Rx ..... Reactor
SAE ..... Site Area Emergency
SAG ..... Severe Accident Guideline
SFP ..... Spent Fuel Pool
SCBA ..... Self-Contained Breathing Apparatus
SPDS ..... Safety Parameter Display System
SRV ..... Safety Relief Valve
SSE ..... Safe Shutdown Earthquake
TAF ..... Top of Active Fuel
TEDE ..... Total Effective Dose Equivalent
UE ..... Unusual Event
USAR ..... Updated Safety Analysis Report
USGS ..... United States Geological Survey
VDC ..... Volts Direct Current
' ..... Feet
" ..... Inches
% ..... Percent
> ..... Greater Than
< ..... Less Than
≥ ..... Greater Than or Equal To
≤ ..... Less Than or Equal To

ATTACHMENT 6 INFORMATION SHEET [INFORMATION USE]

1. PURPOSE

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

1.2 Procedure provides the formal set of threshold conditions necessary to classify an event at CNS into one of the four emergency classifications described in NUREG-0654, NEI 99-01, Revision 6, and the CNS Emergency Plan.©[2]

2. PRECAUTIONS AND LIMITATIONS

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

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

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

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

2.3 EPIP 5.7.1 is considered part of CNS Emergency Plan. Proposed changes must be processed per Procedure 0.29.1.

3. REFERENCES

3.1 DEVELOPMENTAL

3.1.1 NEI 99-01, Revision 6, Methodology for the Development of Emergency Action Levels for Non-Passive Reactors, ADAMS Accession Number ML12326A805.

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

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

3.1.4 10CFR50.72 Immediate Notification Requirements for Operating Nuclear Power Reactors.

3.1.5 10CFR50.73 License Event Report System.

3.1.6 CNS Technical Specifications Table 1.1-1, Modes.

3.1.7 CNS Offsite Dose Assessment Manual (ODAM).

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

3.1.9 CNS Emergency Plan.

3.1.10 Technical Specifications for the Standardized NUHOMS Horizontal Modular Storage System, Certificate of Compliance 1004 Amendment No. 13, Revision 1.

3.1.11 CNS Security and Safeguards Contingency Plan.

3.1.12 Security Procedure 2.4 Patrols.

3.1.13 Administrative Procedure 0.50.5, Outage Shutdown Safety.

3.1.14 CNS Drawing DWG.2.2 (P3-A-45)

3.2 IMPLEMENTING

3.2.1 EPIP 5.7.1, Emergency Classification.

3.2.2 NEI 99-01, Revision 6 to CNS EAL Comparison Matrix.

3.2.3 CNS EAL Matrix.

3.3 MISCELLANEOUS

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

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

3.3.3 ADAMS Accession No. ML15271A299, EPFAQ 2015-004, Fission Barrier Matrix Criteria, Date Accepted 01-Jul-15.

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

3.3.5 ©[2] IR 81-013 (Commitment Number 811217-01-07), Develop Functional Procedure for Accident Classification. Commitment affects Attachment 6, Step 1.2.

3.3.6 ©[4] NRC Document NLS2014035 (Commitment Number NLS2014035-02), Maintain the capability for classifying fuel damage events at the Alert level threshold for CNS at radioactivity levels 300 μCi/g dose equivalent iodine. Commitment affects Attachment 2, Table F-1, on Page 125 and Page 132 Fuel Clad Barrier Criteria FC4. Commitment affects Attachment 2, Fission Product Barrier Degradation, Pages 118 through 119.

ATTACHMENT 7 MATRIX BASIS CROSS-REFERENCE [INFORMATION USE]

CAT	SUB	GE		SAE		ALERT		UE	
A	1	AG1.1	44	AS1.1	38	AA1.1	29	AU1.1	24
		AG1.2	47	AS1.2	41	AA1.2	32	AU1.2	27
		AG1.3	48	AS1.3	42	AA1.3	34		
						AA1.4	36		
	2	AG2.1	59	AS2.1	58	AA2.1	53	AU2.1	50
						AA2.2	55		
						AA2.3	57		
	3					AA3.1	60		
						AA3.2	62		
E	1							EU1.1	116
H	1			HS1.1	171	HA1.1	168	HU1.1	165
	2							HU2.1	173
	3							HU3.1	176
								HU3.2	177
								HU3.3	179
								HU3.4	180
	4							HU4.1	181
								HU4.2	183
								HU4.3	186
								HU4.4	187
	5					HA5.1	188		
	6			HS6.1	192	HA6.1	191		
	7	HG7.1	199	HS7.1	197	HA7.1	195	HU7.1	194
S	1	SG1.1	213	SS1.1	210	SA1.1	206	SU1.1	203
		SG1.2	217						
	2			SS2.1	221				
	3					SA3.1	225	SU3.1	223
	4							SU4.1	228
								SU4.2	230

CAT	SUB	GE		SAE		ALERT		UE	
	5							SU5.1	231
S	6			SS6.1	243	SA6.1	240	SU6.1	233
								SU6.2	237
	7							SU7.1	246
	8					SA8.1	248		
F	1	FG1.1	122	FS1.1	121	FA1.1	120		
C	1	CG1.1	85	CS1.1	77	CA1.1	72	CU1.1	67
		CG1.2	89	CS1.2	79	CA1.2	74	CU1.2	69
				CS1.3	82				
	2					CA2.1	96	CU2.1	93
	3					CA3.1	104	CU3.1	99
								CU3.2	101
	4							CU4.1	107
	5							CU5.1	110
	6					CA6.1	112		

Fission Product Barrier Matrix						
	Fuel Clad Barrier		**Reactor Coolant System Barrier**		**Primary Containment Barrier**	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A	FC1 126	FC2 128	RC S1 135			PC1 148
B			RS C2 138	RCS4 141	PC2 149	
			RC S3 140			
C			RC S5 143		PC3 151	PC5 153
					PC4 152	PC6 154
						PC7 156
D	FC3 131		RC S6 145			PC8 157
	FC4 132					
E					PC9 158	
					PC10 160	
F	FC5 133	FC6 134	RC S7 146	RCS8 147	PC11 161	PC12 162