ENCLOSURE 4

NAPS EAL SCHEME REVISIONS

SUPPORTING DOCUMENTS

Virginia Electric and Power Company
(Dominion Energy Virginia)
North Anna Power Station Units 1 and 2 and ISFSIs

ATTACHMENT 1

NAPS EAL COMPARISON MATRIX DOCUMENT

**Virginia Electric and Power Company
(Dominion Energy Virginia)
North Anna Power Station Units 1 and 2 and ISFSIs**

North Anna Power Station
NEI 99-01, Revision 6
EAL Comparison Matrix

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Introduction

A comparison of the Initiating Conditions (ICs), Mode Applicability and Emergency Action Levels (EALs) in NEI 99-01, Revision 6, Final, "Development of Emergency Action Levels for Non-Passive Reactors", (ADAMS Accession No. ML12326A805), and North Anna Power Station (NAPS) ICs, MODE Applicability and EALs are provided in this document. The results of the comparison are provided in Table 4, NAPS Comparison Matrix.. This document provides a means of assessing NAPS differences and deviations from the NRC endorsed guidance given in NEI 99-01, Revision 6. Discussion of NAPS EAL bases and lists of source document references are given in the NAPS EAL Technical Bases Document. It is, therefore, advisable to reference the NAPS EAL Technical Bases Document for background information while using this document.

Comparison Matrix Format

The ICs and EALs discussed in the NAPS Comparison Matrix are grouped according to NEI 99-01 Recognition Category and presented alphabetically by group. Within each Recognition Category group, the ICs and EALs are listed in tabular format according to the order in which they are given in NEI 99-01, Rev. 6. Generally, each row of the comparison matrix provides the following information:

- NEI IC/Ex. EAL identifier
- NEI IC/Example EAL wording and mode applicability
- NAPS IC/EAL identifier
- NAPS IC/EAL wording and mode applicability
- Justification of any difference or deviation

EAL Wording

NEI 99-01, Section 4.1 recommends the following: "The guidance in NEI 99-01 is not intended to be applied to plants "as-is"; however, developers should attempt to keep their site-specific schemes as close to the generic guidance as possible. The goal is to meet the intent of the generic Initiating Conditions (ICs) and Emergency Action Levels (EALs) within the context of site-specific characteristics – locale, plant design, operating features, terminology, etc. Meeting this goal will result in a shorter and less cumbersome NRC review and approval process, closer alignment with the schemes of other nuclear power plant sites and better positioning to adopt future industry-wide scheme enhancements"

To assist the Station Emergency Manager (SEM), the NAPS EALs have been written in a clear and concise style (to the extent that the differences from the NEI EAL wording could be reasonably documented and justified). This supports timely and accurate classification in the tense atmosphere of an emergency event. The EAL differences introduced to reduce reading burden comprise almost all of the differences justified in this document.

EAL Emphasis Techniques

Due to the width of the table columns and table formatting constraints in this document, line breaks and indentation may differ slightly from the appearance of comparable wording in the source documents. NEI 99-01, Rev. 6, is the source document for the NEI EALs; the NAPS EAL Technical Bases Document is the source document for the NAPS EALs.

Development of the NAPS IC/EAL wording has attempted to minimize inconsistencies and apply sound human factors principles. As a result, differences occur between NEI and NAPS ICs/EALs for these reasons alone. When such difference may infer a technical difference in the associated NEI IC/EAL, the difference is identified and a justification is provided.

The print and paragraph formatting conventions summarized below guide presentation of the NAPS EALs in accordance with the EAL writing criteria. Space restrictions in the EAL table of this document sometimes override this criteria in cases when following the criteria would introduce undesirable complications in the EAL layout.

- Upper case-bold underline print is used for the logic terms **AND**, **OR** and **EITHER**.
- Bold print is also used for certain logic terms, negative terms (**not**, **cannot**, etc.), **any**, **all**.
- Upper case print is reserved for defined terms, acronyms, system abbreviations, logic terms (and, or, etc. when not used as a conjunction), and annunciator window engravings.

- Three or more items in a list are normally introduced with "**Any** of the following..." or "**All** of the following..." Items of the list begin with bullets when a priority or sequence is not inferred.
- The use of **and/or** logic within the same EAL has been avoided when possible. When such logic cannot be avoided, indentation and separation of subordinate contingent phrases is employed.

Global Differences

The differences listed below generally apply throughout the set of EALs and are not repeated in the Justification sections of this document. The global differences do not change the intent of NEI 99-01.

1. The NEI phrase "Notification of Unusual Event" has been abbreviated "NOUE" to reduce EAL-user reading burden.
2. The title "Emergency Director" is replaced with the NAPS-specific title "Station Emergency Director (SEM)"
3. NEI 99-01 IC Example EALs are implemented in separate plant EALs to improve clarity and readability. For example, NEI lists all IC HU3 Example EALs under one IC. The corresponding NAPS EALs appear as unique EALs (e.g., HU3.1 through HU3.4).
4. Operational Condition (MODE) applicability identifiers (numbers/letter) modify the NEI 99-01 mode applicability names as follows: 1 - Power Operation, 2 – Startup, 3 - Hot Standby, 4 - Hot Shutdown, 5 - Cold Shutdown, 6 - Refueling, DEF – Defueled. NEI 99-01 defines Defueled as follows: "All reactor fuel removed from RPV. (Full core off load during refueling or extended outage)."
5. NEI 99-01 uses the terms greater than, less than, greater than or equal to, etc. in the wording of some example EALs. For consistency and to reduce EAL-user reading burden, NAPS has adopted use of boolean symbols in place of the NEI 99-01 text modifiers within the EAL wording.
6. "min." is the standard abbreviation for "minutes" and is used to reduce EAL user reading burden.
7. All ICs and EAL thresholds specifying "thyroid CDE" have been revised to "adult thyroid CDE." The NAPS dose assessment methodology calculates both child and adult thyroid CDE. All effluent based EALs therefore specify "adult thyroid CDE."
8. IC/EAL identification:
   - NEI Recognition Category A, "Abnormal Radiation Levels/ Radiological Effluents," has been changed to Category R, "Abnormal Rad Levels / Rad Effluents." The designator "R" is more intuitively associated with radiation (rad) or radiological events. NEI IC designators beginning with "A" have likewise been changed to "R."
   - NEI Recognition Category S, "System Malfunctions," has been changed to Category M, "System Malfunctions," The designator "M" precludes possible interpretation of "SA" as Site Area Emergency.
   - NEI 99-01 defines the thresholds requiring emergency classification (example EALs) and assigns them to ICs which, in turn, are grouped in "Recognition Categories." NAPS endeavors to optimize the NEI EAL organization and identification scheme to enhance usability of the plant-specific EAL set. To this end, the NAPS IC/EAL scheme includes the following features:
     a. Division of the NEI EAL set into three groups:
        - EALs applicable under **all** plant operating conditions – This group would be reviewed by the EAL-user any time emergency classification is considered.
        - EALs applicable only under <u>hot</u> operating conditions – This group would only be reviewed by the EAL-user when the plant is in Power Operation, Reactor Critical, Hot Shutdown or Intermediate Shutdown mode.
        - EALs applicable only under <u>cold</u> operating conditions – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling or Defueled mode.

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

b. Within each of the above three groups, assignment of EALs to categories/subcategories – Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. Subcategories are used as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The NAPS EAL categories/subcategories and their relationship to NEI Recognition Categories are listed in Table 1.

c. Unique identification of each EAL – Four characters comprise the EAL identifier as illustrated in Figure 1.

Figure 1 – EAL Identifier

EAL Identifier

XXX.X

Category (R, H, E, M, F, C)
Emergency classification (G, S, A, U)
Sequential number within subcategory/classification
Subcategory number (1 if no subcategory)

The first character is a letter associated with the category in which the EAL is located. The second character is a letter associated with the emergency classification level (G for General Emergency, S for Site Area Emergency, A for Alert, and U for Notification of Unusual Event). The third character is a number associated with one or more subcategories within a given category. Subcategories are sequentially numbered beginning with the number "1". If a category does not have a subcategory, this character is assigned the number "1". The fourth character is a number preceded by a period for each EAL within a subcategory. EALs are sequentially numbered within the emergency classification level of a subcategory beginning with the number "1".

The EAL identifier is designed to fulfill the following objectives:

- Uniqueness – The EAL identifier ensures that there can be no confusion over which EAL is driving the need for emergency classification.
- Speed in locating the EAL of concern – When the EALs are displayed in a matrix format, knowledge of the EAL identifier alone can lead the EAL-user to the location of the EAL within the classification matrix. The identifier conveys the category, subcategory and classification level. This assists ERO responders (who may not be in the same facility as the SEM) to find the EAL of concern in a timely manner without the need for a word description of the classification threshold.
- Possible classification upgrade – The category/subcategory/identifier scheme helps the EAL-user find higher emergency classification EALs that may become active if plant conditions worsen.

Table 2 lists the NAPS ICs and EALs that correspond to the NEI ICs/Example EALs when the above EAL/IC organization and identification scheme is implemented.

Differences and Deviations

In accordance NRC Regulatory Issue Summary (RIS) 2003-18, "Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels," Supplements 1 and 2, a "difference" is an EAL change in which the basis scheme guidance differs in wording but agrees in meaning and intent, such that classification of an event would be the same, whether using the basis scheme guidance or the NAPS EAL. A "deviation" is an EAL change in which the basis scheme guidance differs in wording and is altered in meaning or intent, such that classification of the event could be different between the basis scheme guidance and the NAPS proposed EAL.

Administrative changes that do not actually change the textual content are neither differences nor deviations. Likewise, any format change that does not alter the wording of the IC or EAL is not considered a difference or a deviation.

The following are examples of differences:

- Choosing the applicable EAL based upon plant type (i.e., BWR vs. PWR).
- Using a numbering scheme other than that provided in NEI 99-01 that does not change the intent of the overall scheme.
- Where the NEI 99-01 guidance specifically provides an option to not include an EAL, if equipment for the EAL does not exist at NAPS (e.g., automatic real-time dose assessment capability).
- Pulling information from the bases section up to the actual EAL that does not change the intent of the EAL.
- Choosing to state ALL Operating Modes are applicable instead of stating N/A, or listing each mode individually under the Abnormal Rad Level/Radiological Effluent and Hazard and Other Conditions Affecting Plant Safety sections.
- Using synonymous wording (e.g., greater than or equal to vs. at or above, less than or equal vs. at or below, greater than or less than vs. above or below, etc.)
- Adding NAPS equipment/instrument identification and/or noun names to EALs.
- Combining like ICs that are exactly the same but have different operating modes as long as the intent of each IC is maintained and the overall progression of the EAL scheme is not affected.
- Any change to the IC and/or EAL, and/or basis wording, as stated in NEI 99-01, that does not alter the intent of the IC and/or EAL, i.e., the IC and/or EAL continues to:
    - Classify at the correct classification level.
    - Logically integrate with other EALs in the EAL scheme.
    - Ensure that the resulting EAL scheme is complete (i.e., classifies all potential emergency conditions).

The following are examples of deviations:

- Use of altered mode applicability.
- Altering key words or time limits.
- Changing words of physical reference (protected area, safety-related equipment, etc.).
- Eliminating an IC. This includes the removal of an IC from the Fission Product Barrier Degradation category as this impacts the logic of Fission Product Barrier ICs.
- Changing a Fission Product Barrier from a Loss to a Potential Loss or vice-versa.
- Not using NEI 99-01 definitions. The intent is for all NEI 99-01 users to have a standard set of defined terms as delineated in NEI 99-01. Differences due to plant types are permissible (BWR or PWR). Verbatim compliance to the wording of defined terms in NEI 99-01 is not necessary as long as the intent of the defined word is maintained. Use of the wording provided in NEI 99-01 is encouraged since the intent is for all users to have a standard set of delineated terms as defined in NEI 99-01.
- Any change to the IC and/or EAL, and/or basis wording as stated in NEI 99-01 that does alter the intent of the IC and/or EAL (For example, the IC and/or EAL):
    - Does not classify at the classification level consistent with NEI 99-01.
    - Is not logically integrated with other EALs in the EAL scheme.
    - Results in an incomplete EAL scheme (i.e., does not classify all potential emergency conditions).

The "Difference/Deviation Justification" identifies each difference between the NEI 99-01 IC/EAL wording and the NAPS IC/EAL wording. Justification for each difference is then provided. If the difference is determined to be a deviation, a statement is made to that affect and an explanation is provided as to why classification may be different from the NEI 99-01, Rev. 6 IC/EAL and the reason it is acceptable. In all cases, however, the differences and deviations do not change the intent of NEI 99-01. A summary list of NAPS EAL deviations from NEI 99-01, Rev. 6 is provided in Table 3.

Table 1 – NAPS EAL Categories/Subcategories

NAPS EALs		NEI Recognition Category
Category	**Subcategory**	
Group: Any Operating Mode:		
R – Abnormal Rad Levels/Rad Effluent	1 – Radiological Effluent 2 – Irradiated Fuel Event 3 – Area Radiation Levels	Abnormal **R**ad Levels/Radiological Effluent ICs/EALs
H – Hazards 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 – SEM Judgment	**H**azards and Other Conditions Affecting Plant Safety ICs/EALs
E – ISFSI	1 – Confinement Boundary	ISFSI ICs/EALs
Group: Hot Conditions:		
M – System Malfunction	1 – Loss of Emergency 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 – Containment Failure 9 – Hazardous Event Affecting Safety Systems	System **M**alfunction ICs/EALs
F – Fission Product Barrier	None	**F**ission Product Barrier ICs/EALs
Group: Cold Conditions:		
C – Cold Shutdown/Refueling System Malfunction	1 – RCS Level 2 – Loss of Emergency AC Power 3 – RCS Temperature 4 – Loss of Vital DC Power 5 – Loss of Communications 6 - Hazardous Event Affecting Safety Systems	**C**old Shutdown./ Refueling System Malfunction ICs/EALs

Table 2 – NEI / NAPS EAL Identification Cross-Reference

NEI		NAPS	
IC	Example EAL	Category and Subcategory	EAL
AU1	1	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RU1.1 RU1.3
AU1	2	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	N/A
AU1	3	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RU1.2 RU1.4
AU2	1	R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	RU2.1
AA1	1	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RA1.1
AA1	2	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RA1.2
AA1	3	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RA1.3
AA1	4	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RA1.4
AA2	1	R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	RA2.1
AA2	2	R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	RA2.2
AA2	3	R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	RA2.3
AA3	1	R – Abnormal Rad Levels / Rad Effluent, 3 – Area Radiation Levels	RA3.1
AA3	2	R – Abnormal Rad Levels / Rad Effluent, 3 – Area Radiation Levels	RA3.2
AS1	1	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RS1.1
AS1	2	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RS1.2

Table 2 – NEI / NAPS EAL Identification Cross-Reference

NEI		NAPS	
IC	Example EAL	Category and Subcategory	EAL
AS1	3	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RS1.3
AS2	1	R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	RS2.1
AG1	1	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RG1.1
AG1	2	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RG1.2
AG1	3	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RG1.3
AG2	1	R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	RG2.1
CU1	1	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CU1.1
CU1	2	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CU1.2
CU2	1	C – Cold SD/ Refueling System Malfunction, 2 – Loss of AC Power	CU2.1
CU3	1	C – Cold SD/ Refueling System Malfunction, 3 – RCS Temperature	CU3.1
CU3	2	C – Cold SD/ Refueling System Malfunction, 3 – RCS Temperature	CU3.2
CU4	1	C – Cold SD/ Refueling System Malfunction, 4 – Loss of DC Power	CU4.1
CU5	1, 2, 3	C – Cold SD/ Refueling System Malfunction, 5 – Loss of Communications	CU5.1
CA1	1	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CA1.1
CA1	2	C – Cold SD/ Refueling System Malfunction, 1 – RCSV Level	CA1.2
CA2	1	C – Cold SD/ Refueling System Malfunction, 1 – Loss of AC Power	CA2.1
CA3	1, 2	C – Cold SD/ Refueling System Malfunction, 3 – RCS Temperature	CA3.1

Table 2 – NEI / NAPS EAL Identification Cross-Reference

NEI		NAPS	
IC	Example EAL	Category and Subcategory	EAL
CA6	1	C – Cold SD/ Refueling System Malfunction, 6 – Hazardous Event Affecting Safety Systems	CA6.1
CS1	1	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CS1.1
CS1	2	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	N/A
CS1	3	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CS1.2
CG1	1	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CG1.1
CG1	2	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CG1.2
E-HU1	1	E – ISFSI, 1 – Confinement Boundary	EU1.1
FA1	1	F – Fission Product Barrier	FA1.1
FS1	1	F – Fission Product Barrier	FS1.1
FG1	1	F – Fission Product Barrier	FG1.1
HU1	1, 2, 3	H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security	HU1.1
HU2	1	H – Hazards and Other Conditions Affecting Plant Safety, 2 – Seismic Event	HU2.1
HU3	1	H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technology Hazard	HU3.1
HU3	2	H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technology Hazard	HU3.2
HU3	3	H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technology Hazard	HU3.3
HU3	4	H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technology Hazard	HU3.4
HU3	5	H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technology Hazard	N/A

Table 2 – NEI / NAPS EAL Identification Cross-Reference

NEI		NAPS	
IC	Example EAL	Category and Subcategory	EAL
HU4	1	H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire or Explosion	HU4.1
HU4	2	H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire or Explosion	HU4.2
HU4	3	H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire or Explosion	HU4.3
HU4	4	H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire or Explosion	HU4.4
HU7	1	H – Hazards and Other Conditions Affecting Plant Safety, 7 – Judgment	HU7.1
HA1	1, 2	H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security	HA1.1
HA5	1	H – Hazards and Other Conditions Affecting Plant Safety, 5 – Hazardous Gases	HA5.1
HA6	1	H – Hazards and Other Conditions Affecting Plant Safety, 6 – Control Room Evacuation	HA6.1
HA7	1	H – Hazards and Other Conditions Affecting Plant Safety, 7 – Judgment	HA7.1
HS1	1	H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security	HS1.1
HS6	1	H – Hazards and Other Conditions Affecting Plant Safety, 6 – Control Room Evacuation	HS6.1
HS7	1	H – Hazards and Other Conditions Affecting Plant Safety, 7 – Judgment	HS7.1
HG1	1	N/A	N/A
HG7	1	H – Hazards and Other Conditions Affecting Plant Safety, 7 – Judgment	HG7.1
SU1	1	M – System Malfunction, 1 – Loss of AC Power	MU1.1
SU2	1	M – System Malfunction, 3 – Loss of Control Room Indications	MU3.1
SU3	1	M – System Malfunction, 4 – RCS Activity	MU4.1

Table 2 – NEI / NAPS EAL Identification Cross-Reference

NEI		NAPS	
IC	Example EAL	Category and Subcategory	EAL
			MU4.2
SU3	2	M – System Malfunction, 4 – RCS Activity	MU4.3
SU4	1, 2, 3	M – System Malfunction, 5 – RCS Leakage	MU5.1
SU5	1	M – System Malfunction, 6 – RPS Failure	MU6.1
SU5	2	M – System Malfunction, 6 – RPS Failure	MU6.2
SU6	1, 2, 3	M – System Malfunction, 7 –Loss of Communications	MU7.1
SU7	1, 2	M – System Malfunction, 8 – Containment Failure	MU8.1
SA1	1	M – System Malfunction, 1 – Loss of AC Power	MA1.1
SA2	1	M – System Malfunction, 3 – Loss of Control Room Indications	MA3.1
SA5	1	M – System Malfunction, 6 – RPS Failure	MA6.1
SA9	1	M – Hazardous Event Affecting Safety Systems	MA9.1
SS1	1	M – System Malfunction, 1 – Loss of AC Power	MS1.1
SS5	1	M – System Malfunction, 6 – RPS Failure	MS6.1
SS8	1	M – System Malfunction, 2 – Loss of DC Power	MS2.1
SG1	1	M – System Malfunction, 1 – Loss of AC Power	MG1.1
SG8	2	M – System Malfunction, 2 – Loss of DC Power	MG2.1

Table 3 – Summary of Deviations

NEI		NAPS EAL	Description
IC	Example EAL		
AU1	1, 2, 3	RU1.1, RU1.2, RU1.3, RU1.4	Generic IC AU1 has been split to address gaseous and liquid releases separately. The basis for the gaseous UE IC and associated thresholds has been revised to correspond to any unplanned release of gaseous effluent radioactivity to the environment that will result in greater than 1 mrem TEDE. This UE gaseous release criterion is being used consistently at all Dominion Energy nuclear stations (Millstone, North Anna and Surry). The reason this alternative criterion is required is due to the fact that for some effluent gaseous release pathways, the resulting calculated UE threshold following the NEI 99-01 guidance of two times the site specific effluent release limit would result in a UE threshold value greater than the corresponding calculated ALERT threshold based on exceeding 10 mrem TEDE. For the other gaseous release pathways that did not show an incongruent relationship when compared to the ALERT threshold, many showed UE values essentially equivalent to 1 mrem TEDE when applying the guidance in NEI 99-01 of a value set at two times the site specific effluent release limit. The fact that, (1) many of the gaseous release pathway UE values following NEI 99-01 guidance were essentially equivalent to 1 mrem TEDE, (2) application of an alternative definition set at a value of 1 mrem TEDE results in a more limiting value for those release paths that showed incongruent comparison to the corresponding ALERT threshold, and (3) UE criterion set at a value ten (10) times lower than the ALERT threshold provides a logical and consistent escalation between each classification level, provides justification for the UE criterion of 1 mrem TEDE. This single Initiating Condition (IC) definition for gaseous releases at the UE level is being applied to maintain consistency across the Dominion Energy nuclear fleet and to reduce confusion and human error potential if two different (IC) definitions were applied. Due to the fact that there are no ODCM limits on steam safeties or auxiliary feedwater exhausts and the limited ability for these respective radiation monitors to detect low level radioactivity in these steam line configurations, the UE classification thresholds for the steam safeties and auxiliary feedwater exhaust are being labeled N/A (not applicable). **This revised IC and associated thresholds is a deviation from the NEI 99-01**

Table 3 – Summary of Deviations

NEI		NAPS EAL	Description
IC	Example EAL		
			Revision 6 AU1 generic wording and bases but is deemed acceptable consistent with the above justification.
HG1	1	N/A	IC HG1 and associated example EAL is not implemented in the NAPS scheme. There are several other ICs that are redundant with this IC, and are better suited to ensure timely and effective emergency declarations. In addition, the development of new spent fuel pool level EALs, as a result of NRC Order EA-12-051, clarified the intended emergency classification level for spent fuel pool level events. This deviation is justified because: 1. Hostile Action in the Protected Area is bounded by ICs HS1 and HS7. Hostile Action resulting in a loss of physical control is bounded by EAL HG7, as well as any event that may lead to radiological releases to the public in excess of Environmental Protection Agency (EPA) Protective Action Guides (PAGs). a. If, for whatever reason, the Control Room must be evacuated, and control of safety functions (e.g., reactivity control, core cooling, and RCS heat removal) cannot be reestablished, then IC HS6 would apply, as well as IC HS7 if desired by the EAL decision-maker. b. Also, as stated above, any event (including Hostile Action) that could reasonably be expected to have a release exceeding EPA PAGs would be bounded by IC HG7. c. From a Hostile Action perspective, ICs HS1, HS7 and HG7 are appropriate, and therefore, make this part of HG1 redundant and unnecessary. d. From a loss of physical control perspective, ICs HS6, HS7 and HG7 are appropriate, and therefore, make this part of HG1 redundant and unnecessary. 2. Any event which causes a loss of spent fuel pool level will be bounded by ICs AA2, AS2 and AG2, regardless of whether it was based upon a Hostile Action or not, thus making this part of HG1 redundant and unnecessary.

Table 3 – Summary of Deviations

NEI IC	NEI Example EAL	NAPS EAL	Description
			a. An event that leads to a radiological release will be bounded by ICs AU1, AA1, AS1 and AG1. Events that lead to radiological releases in excess of EPA PAGs will be bounded by EALs AG1 and HG7, thus making this part of HG1 redundant and unnecessary. ICs AA2, AS2, AG2, AS1, AG1, HS1, HS6, HS7 and HG7 have been implemented consistent with NEI 99-01, Revision 6 and thus HG1 is adequately bounded as described above. **This exclusion of the generic HG1 guidance is a deviation from the NEI 99-01, Revision 6 generic guidance but is deemed acceptable consistent with endorsed NRC EP FAQ 2015-013.**
HS6	1	HS6.1	Deleted defueled mode applicability. Control of the cited safety functions are not critical for a defueled reactor as there is no energy source in the reactor vessel or RCS. The Mode applicability for the reactivity control safety function has been limited to Modes 1, 2, and 3 (hot operating conditions). In the cold operating modes, adequate shutdown margin exists under all conditions. **This revised mode applicability is a deviation from the NEI 99-01, Revision 6 HS6 generic guidance but is deemed acceptable consistent with endorsed NRC EP FAQ 2015-014.**
CA6 SA9	1 1	CA6.1 MA9.1	The proposed NAPS CA6.1 and MA9.1 wording is intended to ensure that an Alert should be declared only when actual or potential performance issues with SAFETY SYSTEMS have occurred as a result of a hazardous event. The NOUE classification at a minimum. In order to warrant escalation to the Alert classification, the hazardous event must cause indications of degraded performance to one train of a SAFETY SYSTEM with either an indication of degraded performance on the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second SAFETY SYSTEM train, such that the operability or reliability of the second train is a concern. In addition, escalation to the Alert classification should not occur if the damage from the hazardous event is limited

Table 3 – Summary of Deviations

NEI		NAPS EAL	Description
IC	Example EAL		
			to a SAFETY SYSTEM that was inoperable, or out of service, prior to the event occurring. As such, the proposed EALs will reduce the potential of declaring an Alert when events are in progress that do not involve an actual or potential substantial degradation of the level of safety of the plant, (i.e., does not cause significant concern with shutting down or cooling down the plant). EALs CA6.1 and MA9.1 do not directly escalate to a Site Area Emergency or a General Emergency due to a hazardous event. The Fission Product Barrier and/or Abnormal Radiation Levels/Radiological Effluent recognition categories would provide an escalation path to a Site Area Emergency or a General Emergency. The EALs and the Basis sections have been revised to ensure potential escalations from a NOUE to an Alert, due to a hazardous event. This is appropriate as the concern with these EALs is: (1) a hazardous event has occurred, (2) one SAFETY SYSTEM train is having performance issues as a result of the hazardous event, and (3) either the second SAFETY SYSTEM train is having performance issues or the VISIBLE DAMAGE indicates that the second SAFETY SYSTEM train may have operability or reliability issues. The definition of VISIBLE DAMAGE has been revised to reflect the fact that the EALs are based upon SAFETY SYSTEM trains rather than individual components or structures. Note 9 has been added to CA6.1 and MA9.1 as it meets the intent of the EALs, is consistent with other EALs (e.g., EAL HA5.1 which was previously endorsed by the NRC), and ensures that declared emergencies are based upon unplanned events with the potential to pose a radiological risk to the public. Note 10 has been added to CA6.1 and MA9.1 to help reinforce and succinctly capture the more detailed information from the revised basis section related to when conditions would require the declaration of an Alert. CA6.1 and MA9.1 are consistent with NRC FAQ 2016-002 requiring degraded performance or visible damage to more than one safety system train caused by the specified events.

Table 3 – Summary of Deviations

NEI		NAPS EAL	Description
IC	**Example EAL**		
			This revised wording is a deviation from the NEI 99-01, Revision 6 CA6 and SA9 generic wording and bases but is deemed acceptable consistent with endorsed NRC EP FAQ 2016-002.
SG1	1	MG1.1	The proposed NAPS MG1.1 omits the Station Blackout (SBO) coping time threshold. As proposed, the General Emergency classification would be based on a loss of all onsite and offsite AC power to the emergency buses with indications of degraded core cooling. The NAPS SBO analysis and derived coping time was determined in accordance with 10CFR50.63 and Regulatory Guide 1.155. This analysis does not take credit for plant capabilities in place to mitigate the effects of an extended loss of AC power (ELAP). These capabilities were developed and implemented to meet the requirements of NRC Orders EA-12-049 and EA-12-051, and pending regulations in 10 CFR 50.155 (per SECY-16-0142). In accordance with plant EOPs [1(2)-ECA-0.0], operators will declare an ELAP within 60 min. of the loss of all AC power to the emergency buses and direct implementation of FLEX Support Guidelines, including the deployment of dedicated portable equipment and performance of DC load shedding. Even if no AC emergency bus is energized, these actions will maintain or restore core cooling, containment, and spent fuel pool cooling capabilities indefinitely. Therefore, the underlying basis for the generic EAL coping time statement, that power must be restored to an AC emergency bus within a fixed amount of time to avoid a severe challenge to one or more fission product barriers, is not valid for NAPS. Additionally, the omission of the SBO coping time threshold does not remove the attribute of a likely General Emergency declaration prior to meeting the IC FG1 thresholds for ELAP events in which the RCS barrier has not been lost. **This revised wording is a deviation from the NEI 99-01, Revision 6 SG1 generic wording and bases but is deemed appropriate and acceptable.**

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

NEI IC#	NEI IC Wording and Mode Applicability	NAPS IC#(s)	NAPS IC Wording and Mode Applicability	Difference/Deviation Justification
AU1	Release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer. MODE: All	RU1a	Release of liquid radioactivity greater than 2 times the allocated ODCM limits for 60 minutes or longer MODE: All	Generic IC AU1 has been split to address gaseous and liquid releases separately. The NAPS ODCM is the site-specific effluent release controlling document.
		RU1b	Release of gaseous radioactivity resulting in offsite dose greater than 1 mrem TEDE MODE: All	Generic IC AU1 has been split to address gaseous and liquid releases separately. The basis for the gaseous UE IC and associated thresholds has been revised to correspond to any unplanned release of gaseous effluent radioactivity to the environment that will result in greater than 1 mrem TEDE. This UE gaseous release criterion is being used consistently at all operating Dominion Energy nuclear stations (Millstone, North Anna and Surry). The reason this alternative criterion is required is due to the fact that for some effluent gaseous release pathways, the resulting calculated UE threshold following the NEI 99-01 guidance of two times the site specific effluent release limit would result in a UE threshold value greater than the corresponding calculated ALERT threshold based on exceeding 10 mrem TEDE. For the other gaseous release pathways that did not show an incongruent relationship when compared to the ALERT threshold, many showed UE values essentially equivalent to 1 mrem TEDE when applying the guidance in NEI 99-01 of a value set at two times the site specific effluent release limit. The fact that, (1) many of the gaseous release pathway UE values following NEI 99-01 guidance were essentially equivalent to 1 mrem TEDE, (2) application of an alternative definition set at a value of 1 mrem TEDE results in a more limiting value for those release paths that showed incongruent comparison

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

				to the corresponding ALERT threshold, and (3) UE criterion set at a value ten (10) times lower than the ALERT threshold provides a logical and consistent escalation between each classification level, provides justification for the UE criterion of 1 mrem TEDE. This single Initiating Condition (IC) definition for gaseous releases at the UE level is being applied to maintain consistency across the Dominion Energy nuclear fleet and to reduce confusion and human error potential if two different (IC) definitions were applied. Due to the fact that there are no ODCM limits on steam safeties or auxiliary feedwater exhausts and the limited ability for these respective radiation monitors to detect low level radioactivity in these steam line configurations, the UE classification thresholds for the steam safeties and auxiliary feedwater exhaust are being labeled N/A (not applicable). **This revised IC and associated thresholds is a deviation from the NEI 99-01, Revision 6 AU1 generic wording and bases but is deemed acceptable consistent with the above justification.**

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Reading on **ANY** effluent radiation monitor greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer: (site-specific monitor list and threshold values corresponding to 2 times the controlling	RU1.1	Reading on SW-RM-130(230) CW Discharge Tunnel radiation monitor > 2 x the "Hi-Hi" setpoint for ≥60 min. (Notes 1, 2, 3)	The NEI phrase "…effluent radiation monitor greater than 2 times the (site-specific effluent release controlling document) has been replaced with "Reading on SW-RM-130(230) CW Discharge Tunnel radiation monitor > 2 x the "Hi-Hi" setpoint ". Consistent with the above justification, liquid and gaseous effluent thresholds have been split. The CW Discharge Tunnel radiation monitor is the liquid release pathway not associated with discharge permits.

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

	document limits)	RU1.3	Reading on **any** Table R-1 effluent radiation monitor > column "NOUE" for ≥60 min. (Notes 1, 2, 3)	The NEI phrase "…effluent radiation monitor greater than 2 times the (site-specific effluent release controlling document)" has been replaced with "…**any** Table R-1 effluent radiation monitor > column "NOUE". NOUE thresholds for all NAPS continuously monitored gaseous release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL user. The values shown in Table R-1 column "NOUE", consistent with the revised IC bases, corresponds to releases resulting in a 1 mrem dose at the site boundary for a 1-hour release.
2	Reading on **ANY** effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.	N/A	N/A	NAPS does not establish radiation monitor setpoints for liquid batch releases.
3	Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer.	RU1.2	Sample analysis for a liquid release indicates a concentration or release rate > 2 x the allocated ODCM limits for ≥60 min. (Notes 1, 2)	The NAPS ODCM is the site-specific effluent release controlling document.
		RU1.4	Sample analysis for a gaseous release indicates a concentration or release rate > 2 x the allocated ODCM limits for ≥60 min. (Notes 1, 2)	The NAPS ODCM is the site-specific effluent release controlling document.
Notes	• The Emergency Director should declare the Unusual Event promptly upon determining that 60 minutes	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

	has been exceeded, or will likely be exceeded. • If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 60 minutes. • If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.		will likely be exceeded. Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit. Note 3: If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording. None

Table R-1 Gaseous Effluent Monitor Classification Thresholds

Release Point & Monitor	GE	SAE	Alert	NOUE
Vent Stack A VG-RI-179-1 or 2	2.6E+08 µCi/sec	2.6E+07 µCi/sec	2.6E+06 µCi/sec	2.6E+05 µCi/sec
Vent Stack B VG-RI-180-1 or 2	2.0E+08 µCi/sec	2.0E+07 µCi/sec	2.0E+06 µCi/sec	2.0E+05 µCi/sec
Process Vent GW-RI-178-1 or 2	3.5E+08 µCi/sec	3.5E+07 µCi/sec	3.5E+06 µCi/sec	3.5E+05 µCi/sec
Main Steam Line MS-RI-170 (270) MS-RI-171 (271) MS-RI-172 (272)	1.3E+03 mR/hr	1.3E+02 mR/hr	1.3E+01 mR/hr	N/A
TD AFW Pump EXH MS-RI-176 (276)	6.0E+01 mR/hr	6.0E +00 mR/hr	6.0E-01 mR/hr	N/A

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

NEI IC#	NEI IC Wording and Mode Applicability	NAPS IC#(s)	NAPS IC Wording and Mode Applicability	Difference/Deviation Justification
AU2	UNPLANNED loss of water level above irradiated fuel. MODE: All	RU2	UNPLANNED loss of water level above irradiated fuel MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following: (site-specific level indications). **AND** b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors. (site-specific list of area radiation monitors)	RU2.1	UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **any** of the following: • Spent Fuel Pit Lo Level (1E-C6) alarm • Report of dropping level in refueling cavity or SFP • Loss of SFP Cooling suction flow **AND** UNPLANNED rise in corresponding area radiation levels as indicated by **any** of the following radiation monitors: • RM-RMS-152 New Fuel Storage Area • RM-RMS-153 Fuel Pit Bridge • RM-RMS-162 (262) Manipulator Crane Area	Site-specific level indications incorporated. Site-specific area radiation monitors incorporated. Added the word "...corresponding..." to reinforce the cause (water level decrease) and effect (area radiation levels) intent of this EAL.

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

			(Refueling Mode) • RM-RMS-163 (263) Reactor Containment Area	

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

NEI IC#	NEI IC Wording and Mode Applicability	NAPS IC#(s)	NAPS IC Wording and Mode Applicability	Difference/Deviation Justification
AA2	Significant lowering of water level above, or damage to, irradiated fuel. MODE: All	RA2	Significant lowering of water level above, or damage to, irradiated fuel MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Uncovery of irradiated fuel in the REFUELING PATHWAY.	RA2.1	IMMINENT uncovery of irradiated fuel in the REFUELING PATHWAY	Added the term "IMMINENT" consistent with the generic bases.
2	Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by **ANY** of the following radiation monitors: (site-specific listing of radiation monitors, and the associated readings, setpoints and/or alarms)	RA2.2	Damage to irradiated fuel resulting in a release of radioactivity **AND EITHER:** • VALID Hi-Hi alarm on **any** of the following radiation monitors: o RM-RMS-152 New Fuel Storage Area o RM-RMS-153 Fuel Pit Bridge o RM-RMS-162 (262) Manipulator Crane Area (Refueling Mode) o RM-RMS-163 (263) Reactor Containment Area	Deleted the words "…from the fuel…" as that is implied by the determination that irradiated fuel has been damaged. Site-specific list of radiation monitors are incorporated. Valid radiation monitor Hi-Hi alarms specified.

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

			o RM-RMS-159 (259) Containment Particulate o RM-RMS-160 (260) Containment Area Gas • VALID Hi alarm on VG-RI-180-1 Vent Stack B Normal Range	
3	Lowering of spent fuel pool level to (site-specific Level 2 value). [*See Developer Notes*]	RA2.3	Lowering of spent fuel pool level to 10 ft. (Level 2) on 1-FC-LI-105-1, 2 or 2A Spent Fuel Pit Wide Range Level	For NAPS, Level 2, which corresponds to 10 ft. above the top of the fuel racks in the SFP, is an indicated level of 10 ft. on 1-FC-LI-105-1, 2 or 2A Spent Fuel Pit Wide Range Level.

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
AA3	Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown MODE: All	RA3	Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown All (except RA3.2) RA3.2 - MODE: 1 – Power Operations, 2 – Startup, 3 – Hot Shutdown, 4 – Hot Shutdown	Limited mode applicability of RA3.2 specified in Table R-2.

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Dose rate greater than 15 mR/hr in **ANY** of the following areas: • Control Room • Central Alarm Station • (other site-specific areas/rooms)	RA3.1	Dose rates > 15 mR/hr in **EITHER** of the following: • Control Room • Central Alarm Station (by survey)	No other site-specific areas requiring continuous occupancy exist at NAPS. CAS does not have permanently installed area radiation monitoring so dose rates must be assesses by survey.
2	An UNPLANNED event results in radiation levels that prohibit or impede access to any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode applicability identified)	RA3.2	An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 room or area (Note 5)	The site-specific list of plant rooms or areas with entry-related mode applicability are tabularized in Tables R-2.

Table 4 – NAPS Comparison Matrix				
Category A: Abnormal Rad Levels / Radiological Effluent				
Note	If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.	N/A	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.	None

Table R-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode**
Aux. Building El 274'	1, 2, 3, 4
Instrument Rack Rooms	4
Cable Vault & Tunnels	

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
AS1	Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE MODE: All	RS1	Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem adult thyroid CDE MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	RS1.1	Reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥15 min. (Notes 1, 2, 3, 4)	The NAPS radiation monitors that detect radioactivity effluent release to the environment are listed in Table R-1. NOUE, Alert, SAE and GE thresholds for all NAPS continuously monitored gaseous and liquid release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL-user.
2	Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond (site-specific dose receptor point)	RS1.2	Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem adult thyroid CDE at or beyond the SITE BOUNDARY (Note 4)	The site boundary is the site-specific receptor point.
3	Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point): • Closed window dose rates	RS1.3	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 ≥	The site boundary is the site-specific field survey receptor point.

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

	greater than 100 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.		60 min. • Analyses of field survey samples indicate adult thyroid CDE > 500 mrem for 60 min. of inhalation. (Notes 1, 2)	
Notes	• The Emergency Director should declare the Site Area Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded. • If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes. • If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes. • The pre-calculated effluent monitor values presented in EAL #1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.		Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit. Note 3: If the effluent flow past an effluent monitor is known to have stopped 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 RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording. The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording. None Incorporated site-specific EAL numbers associated with generic EAL#1.

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

			available.	

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
AS2	Spent fuel pool level at (site-specific Level 3 description) MODE: All	RS2	Spent fuel pool level at the top of the fuel racks MODE: All	Top of the fuel racks is the site-specific Level 3 description.

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Lowering of spent fuel pool level to (site-specific Level 3 value)	RS2.1	Lowering of spent fuel pool level to 1 ft. (Level 3) on 1-FC-LI-105-1, 2 or 2A Spent Fuel Pit Wide Range Level	For NAPS, Level 3, which corresponds to the top of the fuel racks in the SFP, is 1 ft. on 1-FC-LI-105-1, 2 or 2A Spent Fuel Pit Wide Range Level

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
AG1	Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE. MODE: All	RG1	Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem adult thyroid CDE MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	RG1.1	Reading on **any** Table R-1 effluent radiation monitor > column "GE" for ≥15 min. (Notes 1, 2, 3, 4)	The NAPS radiation monitors that detect radioactivity effluent release to the environment are listed in Tables R-1. NOUE, Alert, SAE and GE thresholds for all NAPS continuously monitored gaseous and liquid release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL-user.
2	Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond (site-specific dose receptor point).	RG1.2	Dose assessment using actual meteorology indicates doses > 1,000 mrem TEDE or 5,000 mrem adult thyroid CDE at or beyond the SITE BOUNDARY (Note 4)	The site boundary is the site-specific receptor point.
3	Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point): • Closed window dose rates greater than 1,000 mR/hr	RG1.3	Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates > 1,000 mR/hr expected to	The site boundary is the site-specific field survey receptor point.

Table 4 – NAPS Comparison Matrix				
Category A: Abnormal Rad Levels / Radiological Effluent				
	expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.		continue for ≥60 min. • Analyses of field survey samples indicate adult thyroid CDE > 5,000 mrem for 60 min. of inhalation. (Notes 1, 2)	
Notes	• The Emergency Director should declare the Site Area Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.		Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.
	• If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.		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.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.
	• 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 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 for classification purposes.	None
	• The pre-calculated effluent monitor values presented in EAL #1 should be used for emergency classification assessments until the results from a dose assessment		Note 4: The pre-calculated	Incorporated site-specific EAL numbers associated with generic

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

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

Table 4 – NAPS Comparison Matrix

Category A: Abnormal Rad Levels / Radiological Effluent

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
AG2	Spent fuel pool level cannot be restored to at least (site-specific Level 3 description) for 60 minutes or longer MODE: All	RG2	Spent fuel pool level **cannot** be restored to at least the top of the fuel racks for 60 minutes or longer MODE: All	Top of the fuel racks is the site-specific Level 3 description.

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Spent fuel pool level cannot be restored to at least (site-specific Level 3 value) for 60 minutes or longer	RG2.1	Spent fuel pool level **cannot** be restored to at least 1 ft. (Level 3) on 1-FC-LI-105-1, 2 or 2A Spent Fuel Pit Wide Range Level for ≥60 min. (Note 1)	For NAPS, Level 3, which corresponds to the top of the fuel racks in the SFP, is 1 ft. 1 ft. indicated is the lower range of the SFP level instrument, therefore an indication > 1 ft. is required.
Note	The Emergency Director should declare the General Emergency promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table 4 – NAPS Comparison Matrix

Category C: Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
CU1	UNPLANNED loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory for 15 minutes or longer. MODE: Cold Shutdown, Refueling	CU1	UNPLANNED loss of RCS inventory MODE: 5 - Cold Shutdown, 6 - Refueling	Deleted the words "…for 15 minutes or longer" as the 15 minute criteria only applies to EAL #1

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	UNPLANNED loss of reactor coolant results in (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level less than a required lower limit for 15 minutes or longer.	CU1.1	UNPLANNED loss of reactor coolant results in RCS water level < a required lower limit for ≥15 min. (Note 1)	None
2	a. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level cannot be monitored. **AND** b. UNPLANNED increase in (site-specific sump and/or tank) levels.	CU1.2	RCS water level **cannot** be monitored **AND EITHER** • UNPLANNED increase in **any** Table C-1 sump or tank level due to a loss of RCS inventory • Visual observation of UNISOLABLE RCS leakage	Added the words "…due to loss of RCS inventory" to be consistent with the IC wording. The Table C-1 sumps & tanks are the site-specific applicable sumps and tanks. Although "Visual observation…" is neither a sump nor tank, it is included in order to implement the intent of the NEI basis which states: "…operators may determine that an inventory loss is occurring by observing changes…"

Table 4 – NAPS Comparison Matrix				
Category C: Cold Shutdown / Refueling System Malfunction				
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table C-1 Sumps/Tanks
• Reactor Containment Sump • Pressurizer Relief Tank (PRT) • Primary Drain Transfer Tank (PDTT) • Component Cooling (CC) Surge Tank • Refueling Water Storage Tank (RWST)

Table 4 – NAPS Comparison Matrix

Category C: Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
CU2	Loss of all but one AC power source to emergency buses for 15 minutes or longer. MODE: Cold Shutdown, Refueling, Defueled	CU2	Loss of **all but one** AC power source to emergency buses for 15 minutes or longer. MODE: 5 - Cold Shutdown, 6 - Refueling, D - Defueled	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. AC power capability to (site-specific emergency buses) is reduced to a single power source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS.	CU2.1	AC power capability, Table C-4, to Unit 1(2) 4160V emergency buses H and J 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	4160V emergency buses H and J are the NAPS-specific emergency buses. Table C-4 provides a consolidated list of AC power sources credited for this EAL.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table C-4 AC Power Sources
Offsite: Unit 1 • Transfer Bus D • Transfer Bus F • Station Bus 1B • Station Bus 2B Unit 2 • Transfer Bus E • Transfer Bus F • Station Bus 2C • Station Bus 1A **Onsite:** • 1(2)H EDG • 1(2)J EDG • AAC (SBO) Diesel Generator (if already aligned)

Table 4 – NAPS Comparison Matrix

Category C: Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
CU3	UNPLANNED increase in RCS temperature MODE: Cold Shutdown, Refueling	CU3	UNPLANNED increase in RCS temperature MODE: 5 - Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit)	CU3.1	UNPLANNED increase in RCS temperature to > 200°F	200°F is the site-specific Tech. Spec. cold shutdown temperature limit.
2	Loss of **ALL** RCS temperature and (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level indication for 15 minutes or longer.	CU3.2	Loss of **all** RCS temperature and RCS water level indication for ≥ 15 min. (Note 1)	None
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table 4 – NAPS Comparison Matrix

Category C: Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
CU4	Loss of Vital DC power for 15 minutes or longer. MODE: Cold Shutdown, Refueling	CU4	Loss of vital DC power for 15 minutes or longer. MODE 5 - Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Indicated voltage is less than (site-specific bus voltage value) on required Vital DC buses for 15 minutes or longer.	CU4.1	Indicated voltage is < 105 VDC on **required** vital 125 VDC battery buses for ≥15 min. (Note 1)	The specified bus voltage indications are the minimum voltage requirements for operability of the 125 VDC buses. Vital 125 VDC battery buses are the vital DC buses credited for the EAL.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table 4 – NAPS Comparison Matrix

Category C: Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
CU5	Loss of all onsite or offsite communications capabilities. MODE: Cold Shutdown, Refueling, Defueled	CU5	Loss of **all** onsite or offsite communications capabilities. MODE: 5 - Cold Shutdown, 6 - Refueling, D - Defueled	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Loss of **ALL** of the following onsite communication methods: (site specific list of communications methods)	CU5.1	Loss of **all** Table C-6 onsite communication methods **OR** Loss of **all** Table C-6 State and local agency communication methods **OR** Loss of **all** Table C-6 NRC communication methods	Example EALs #1, 2 and 3 have been combined into a single EAL for simplification of presentation. Table C-6 provides a site-specific list of onsite, offsite (ORO) and NRC communications methods.
2	Loss of **ALL** of the following ORO communications methods: (site specific list of communications methods)			
3	Loss of **ALL** of the following NRC communications methods: (site specific list of communications methods)			

Table C-6 Communication Methods

System	Onsite	State/ Local	NRC
Radio Communications System	X		
Public Address and Intercom System	X		
Private Branch Telephone Exchange (PBX)	X	X	X
Sound Powered Telephone System	X		
Commercial Telephone System		X	X
Automatic Ring Downs (SONET Ring)		X	
Instaphone Loop		X	
Dedicated NRC Communications			X

Table 4 – NAPS Comparison Matrix

Category C: Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
CA1	Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory MODE: Cold Shutdown, Refueling	CA1	Significant loss of RCS inventory MODE: 5 - Cold Shutdown, 6 - Refueling	Added the word "Significant…" to differentiate the Alert loss of RCS inventory IC from the NOUE IC which is "Unplanned loss of RCS inventory."

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory as indicated by level less than (site-specific level).	CA1.1	RCS level < minimum required for continued RHR pump operation	The classification threshold is based on the lowest RCS level that supports continued decay heat removal pump (RHR) operations per procedure.
2	a. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level cannot be monitored for 15 minutes or longer **AND** b. UNPLANNED increase in (site-specific sump and/or tank) levels due to a loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory.	CA1.2	RCS water level **cannot** be monitored for ≥15 min. (Note 1) **<u>AND EITHER</u>** • UNPLANNED increase in **any** Table C-1 sump or tank level due to a loss of RCS inventory • Visual observation of UNISOLABLE RCS leakage	The Table C-1 sumps/tanks are the site-specific applicable sumps and tanks. Although "Visual observation…" is neither a sump nor tank, it is included in order to implement the intent of the NEI basis which states: "…operators may determine that an inventory loss is occurring by observing changes…"
Note	The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be	N/A	Note 1: The SEM should declare the event promptly upon determining that the	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table 4 – NAPS Comparison Matrix

Category C: Cold Shutdown / Refueling System Malfunction

	exceeded		time limit has been exceeded, or will likely be exceeded.	

Table 4 – NAPS Comparison Matrix

Category C: Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
CA2	Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer MODE: Cold Shutdown, Refueling, Defueled	CA2	Loss of **all** offsite and **all** onsite AC power to emergency buses for 15 minutes or longer. MODE: 5 - Cold Shutdown, 6 - Refueling, DEF - Defueled	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Loss of **ALL** offsite and **ALL** onsite AC Power to (site-specific emergency buses) for 15 minutes or longer.	CA2.1	Loss of **all** offsite and **all** onsite AC power to Unit 1(2) 4160V emergency buses H and J for ≥15 min. (Note 1)	4160V emergency buses H and J are the NAPS-specific emergency buses.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table 4 – NAPS Comparison Matrix

Category C: Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
CA3	Inability to maintain the plant in cold shutdown. MODE: Cold Shutdown, Refueling	CA3	Inability to maintain plant in cold shutdown. MODE: 5 - Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit) for greater than the duration specified in the following table.	CA3.1	UNPLANNED increase in RCS temperature to > 200°F for > Table C-5 duration (Notes 1, 12) **OR** UNPLANNED RCS pressure increase > 10 psi (does not apply to solid plant conditions)	Example EALs #1 and #2 have been combined into a single EAL as EAL #2 is the alternative threshold based on a loss of RCS temperature indication. 200°F is the site-specific Tech. Spec. cold shutdown temperature limit. Table C-5 is the site-specific implementation of the generic RCS Reheat Duration Threshold table. 10 psi is the site-specific RCS pressure increase readable by Control Room indications.
2	UNPLANNED RCS pressure increase greater than (site-specific pressure reading). (This EAL does not apply during water-solid plant conditions. [*PWR*])			
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table 4 – NAPS Comparison Matrix				
Category C: Cold Shutdown / Refueling System Malfunction				
			be exceeded.	
N/A	N/A	N/A	Note 12: If an RCS heat removal system is in operation within the applicable Table C-5 heat-up duration and RCS temperature is being reduced, the EAL is **not** applicable.	Added Note 12 consistent with the asterisk note provided in the generic RCS Heat-up Duration Threshold table.

Table: RCS Heat-up Duration Thresholds		
RCS Status	**Containment Closure Status**	**Heat-up Duration**
Intact (but not at reduced inventory [*PWR*])	Not applicable	60 minutes*
Not intact (or at reduced inventory [*PWR*])	Established	20 minutes*
	Not Established	0 minutes
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

Table C-5 RCS Heat-up Duration Thresholds

RCS Status	CONTAINMENT CLOSURE Status	Heat-up Duration
Intact **<u>AND</u> not** reduced/decreased inventory		60 min.
Not intact **<u>OR</u>** reduced/decreased inventory	Established	20 min.
	Not established	0 min.

Table 4 – NAPS Comparison Matrix

Category C: Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
CA6	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. MODE: Cold Shutdown, Refueling	CA6	Hazardous event affecting SAFETY SYSTEMS needed for the current operating mode. MODE: 5 - Cold Shutdown, 6 - Refueling	Revised wording from "…affecting a SAFETY SYSTEM…" to read "…affecting SAFETY SYSTEMS…" to align with changes made consistent with NRC EP FAQ 2016-002.

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. The occurrence of **ANY** of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • (site-specific hazards) • Other events with similar hazard characteristics as determined by the Shift Manager **AND** b. **EITHER** of the following: 1. Event damage has	CA6.1	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 second train of the SAFETY SYSTEM needed for the current operating mode • Event damage has resulted in VISIBLE	The hazardous events have been tabularized in Table C-7. The proposed NAPS CA6.1 and SA9.1 wording is intended to ensure that an Alert should be declared only when actual or potential performance issues with SAFETY SYSTEMS have occurred as a result of a hazardous event. The occurrence of a hazardous event will result in an NOUE classification at a minimum. In order to warrant escalation to the Alert classification, the hazardous event should cause indications of degraded performance to one train of a SAFETY SYSTEM with either indications of degraded performance on the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second SAFETY SYSTEM train, such that the operability or reliability of the second train is a concern. In addition, escalation to the Alert classification should not occur if the damage from the hazardous event is limited to a SAFETY SYSTEM that was inoperable, or out of service, prior to the event occurring. As such, the proposed EALs will reduce the potential of declaring an Alert when events are in progress that do not involve an actual or potential substantial degradation of the level of safety of the plant, (i.e., does not cause significant concern with shutting down or cooling down the plant).

	caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode. **OR** 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.		DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode (Notes 9, 10)	EALs CA6.1 and SA9.1 do not directly escalate to a Site Area Emergency or a General Emergency due to a hazardous event. The Fission Product Barrier and/or Abnormal Radiation Levels/Radiological Effluent recognition categories would provide an escalation path to a Site Area Emergency or a General Emergency. The EALs and the Basis sections have been revised to ensure potential escalations from a NOUE to an Alert, due to a hazardous event, is appropriate as the concern with these EALs is: (1) a hazardous event has occurred, (2) one SAFETY SYSTEM train is having performance issues as a result of the hazardous event, and (3) either the second SAFETY SYSTEM train is having performance issues or the VISIBLE DAMAGE is enough to indicate that the second SAFETY SYSTEM train may have operability or reliability issues. The definition for VISIBLE DAMAGE has been revised to reflect the fact that the EALs are based upon SAFETY SYSTEM trains rather than individual components or structures. Note 9 has been added to CA6.1 and SA9.1 as it meets the intent of the EALs, is consistent with other EALs (e.g., EAL HA5.1 which was previously endorsed by the NRC), and ensures that declared emergencies are based upon unplanned events with the potential to pose a radiological risk to the public. Note 10 has been added to CA6.1 and SA9.1 to help reinforce and succinctly capture the more detailed information from the revised basis section related to when conditions would require the declaration of an Alert. CA6.1 and SA9.1 are consistent with NRC FAQ 2016-002 requiring degraded performance or visible damage to more than one safety system train caused by the specified events. **This revised wording is a deviation from the NEI 99-01, Revision 6 CA6 and SA9 generic wording and bases but is deemed acceptable consistent with endorsed NRC EP FAQ 2016-002.**

N/A	N/A	N/A	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.	Added Note 9 consistent with the recommendation of NRC EP FAQ 2016-002.
			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.	Added Note 10 consistent with the recommendation of NRC EP FAQ 2016-002.

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/SEM

Table 4 – NAPS Comparison Matrix

Category C: Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
CS1	Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory affecting core decay heat removal capability. MODE: Cold Shutdown, Refueling	CS1	Loss of RCS inventory affecting core decay heat removal capability MODE: 5 - Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. CONTAINMENT CLOSURE not established. **AND** b. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level less than (site-specific level).	CS1.1	With CONTAINMENT CLOSURE **not** established, **any** confirmed loss of inventory indication, Table C-2, with RVLIS full range < 62%	Six inches below the elevation of the bottom of the RCS hot leg penetration can be monitored only by RVLIS full range (62%). Other level monitoring instruments are offscale low when level is below the elevation of the RCS loop hot leg penetration. Table C-2 provides a list of confirmatory indicators for RCS inventory loss. Due to the variability of accuracy and usability of RVLIS while in Cold Shutdown or Refueling Mode, the use of RVLIS for emergency classification purposes is contingent on one or more of the listed confirmatory indications.
2	a. CONTAINMENT CLOSURE established. **AND** b. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level less than (site-specific level).	CS1.2	With CONTAINMENT CLOSURE established, **any** confirmed loss of inventory indication, Table C-2, with RVLIS full range < 61%	This level drop can only be remotely monitored by Reactor Vessel Level Instrumentation System (RVLIS). When Reactor Vessel water level drops below RVLIS full range setpoint of 61%, core uncovery is about to occur. Table C-2 provides a list of confirmatory indicators for RCS inventory loss. Due to the variability of accuracy and usability of RVLIS while in Cold Shutdown or Refueling Mode, the use of RVLIS for emergency classification purposes is contingent on one or more of the listed

Table 4 – NAPS Comparison Matrix

Category C: Cold Shutdown / Refueling System Malfunction

				confirmatory indications.
3	a. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level cannot be monitored for 30 minutes or longer. **AND** b. Core uncovery is indicated by **ANY** of the following: • (Site-specific radiation monitor) reading greater than (site-specific value) • Erratic source range monitor indication [*PWR*] • UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery • (Other site-specific indications)	CS1.3	RCS level **cannot** be monitored for ≥30 min. (Note 1) **<u>AND</u>** Core uncovery is indicated by **any** of the following: • UNPLANNED increase in **any** Table C-1 sump or tank level of sufficient magnitude to indicate core uncovery • Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery • **Any** containment area radiation monitor reading > 3 R/hr (Refueling Mode) • Erratic source range monitor indications	Site-specific applicable sumps and tanks are listed in Table C-1 to improve the readability of the EAL. Although "Visual observation…" is neither a sump nor tank, it is included in order to implement the intent of the NEI basis which states: "…operators may determine that an inventory loss is occurring by observing changes…" In the Refueling mode, as water level in the reactor vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in on-scale indications of > 3 R/hr on containment area radiation monitors. No other site-specific indications of core uncovery have been identified for NAPS.
Note	The Emergency Director should declare the Site Area Emergency promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table C-2 Inventory Loss Confirmatory Indications
• In service Standpipe and Ultrasonic level bottomed out • Decreasing RVLIS level trend • RHR pump amp fluctuations

Table 4 – NAPS Comparison Matrix

Category C: Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
CG1	Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory affecting fuel clad integrity with containment challenged MODE: Cold Shutdown, Refueling	CG1	Loss of RCS inventory affecting fuel clad integrity with containment challenged MODE: 5 - Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level less than (site-specific level) for 30 minutes or longer. **AND** b. **ANY** indication from the Containment Challenge Table (see below).	CG1.1	**Any** confirmed loss of inventory indication, Table C-2, with RVLIS full range < 61% for ≥30 min. (Note 1) **AND** **Any** Containment Challenge indication, Table C-3	This level drop can only be remotely monitored by Reactor Vessel Level Instrumentation System (RVLIS). When Reactor Vessel water level drops below RVLIS full range setpoint of 61%, core uncovery is about to occur. Table C-2 provides a list of confirmatory indicators for RCS inventory loss. Due to the variability of accuracy and usability of RVLIS while in Cold Shutdown or Refueling Mode, the use of RVLIS for emergency classification purposes is contingent on one or more of the listed confirmatory indications.
2	a. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level cannot be monitored for 30 minutes or longer. **AND** b. Core uncovery is indicated by **ANY** of the following:	CG1.2	RCS level **cannot** be monitored for ≥30 min. (Note 1) **AND** Core uncovery is indicated by **any** of the following: • UNPLANNED increase in **any** Table C-1 sump or tank	Site-specific applicable sumps and tanks are listed in Table C-1 to improve the readability of the EAL. Although "Visual observation…" is neither a sump nor tank, it is included in order to implement the intent of the NEI basis which states: "…operators may determine that an inventory loss is occurring by observing changes…" In the Refueling mode, as water level in the reactor vessel lowers, the dose rate above the core will rise. The dose rate due to this

Table 4 – NAPS Comparison Matrix				
Category C: Cold Shutdown / Refueling System Malfunction				
	• (Site-specific radiation monitor) reading greater than (site-specific value) • Erratic source range monitor indication [*PWR*] • UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery • (Other site-specific indications) **AND** c. **ANY** indication from the Containment Challenge Table (see below).		level of sufficient magnitude to indicate core uncovery • Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery • **Any** containment area radiation monitor reading > 3 R/hr (Refueling Mode) • Erratic source range monitor indications **AND** **Any** Containment Challenge indication, Table C-3	core shine should result in on-scale indications of > 3 R/hr on containment area radiation monitors. No other site-specific indications of core uncovery have been identified for NAPS. 4% hydrogen concentration in the presence of oxygen is the minimum necessary to support a hydrogen explosion.
Note	The Emergency Director should declare the General Emergency promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-min. time limit, declaration of a General Emergency is **not** required.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording. Note 6 implements the asterisked note associated with the Containment Closure requirement.

Containment Challenge Table
• CONTAINMENT CLOSURE not established* • (Explosive mixture) exists inside containment • UNPLANNED increase in containment pressure • Secondary containment radiation monitor reading above (site-specific value) [*BWR*]

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

Table C-3 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • CTMT hydrogen concentration ≥4% • UNPLANNED increase in CTMT pressure

Table 4 – NAPS Comparison Matrix

Category D: Permanently Defueled Station Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
PD-AU1 PD-AU2 PD-SU1 PD-HU1 PD-HU2 PD-HU3 PD-AA1 PD-AA2 PD-HA1 PD-HA3	Recognition Category D Permanently Defueled Station	N/A	N/A	NEI Recognition Category PD ICs and EALs are applicable only to permanently defueled stations. NAPS is not a defueled station.

Table 4 – NAPS Comparison Matrix

Category E: Spent Fuel Storage installation

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
E-HU1	Damage to a loaded cask CONFINEMENT BOUNDARY MODE: All	EU1	Damage to a loaded cask CONFINEMENT BOUNDARY MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than (2 times the site-specific cask specific technical specification allowable radiation level) on the surface of the spent fuel cask.	EU1.1	Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask > **any** Table E-1 limit	The Table E-1 specified EAL threshold values correspond to 2 times the Sealed Surface Storage Cask (TN-32/TN-32B) or Horizontal Storage Module (HSM-H) external surface dose rate limits. NAPS utilizes the following dry cask storage systems: • Transnuclear TN-32 • Transnuclear TN-32B HBU • NUHOMS HD System (32PTH DSC/HSM-H)

Table E-1 ISFSI Cask Surface Dose Rate Limits

TN-32	TN-32B HBU	HSM-H
• 116 mrem/hr (neutron + gamma) average on top of the cask • 436 mrem/hr (neutron + gamma) average on the side of the cask	• 192 mrem/hr (neutron + gamma) average on top of the cask • 436 mrem/hr (neutron + gamma) average on the side of the cask	• 1,600 mrem/hr at the front bird screen • 4 mrem/hr at the door centerline • 4 mrem/hr at the end shield wall exterior

Table 4 – NAPS Comparison Matrix

Category F: Fission Product Barrier Degradation

Fuel Clad Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI Threshold Wording	NAPS FPB #(s)	NAPS FPB Wording	Difference Justification
FC Loss 1	**RCS or SG Tube Leakage** Not Applicable	N/A	N/A	N/A
FC Loss 2	**Inadequate Heat Removal** A. Core exit thermocouple readings greater than (site-specific temperature value).	FC Loss B.1	1. Core Cooling-RED Path conditions met	Consistent with the generic developers note options CSFST Core Cooling Red Path is used in lieu of CET temperatures.
FC Loss 3	**RCS Activity/CMNT Rad** A. Containment radiation monitor reading greater than (site-specific value) **OR** B. (Site-specific indications that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131)	FC Loss C.2	2. CTMT high range radiation monitor RM-RMS-165/166(265/266) reading > Table F-2 column Fuel Clad Loss	Monitors RM-RMS-165/166(265/266) are the containment high range area radiation monitors. The threshold values specified in Table F-2 have been calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with approximately 5% fuel clad damage.
		FC Loss C.3	3. Coolant activity > 300 µCi/gm DEI-131	None
		FC Loss C.4	4. Dose rate at 1 ft. from an unpressurized RCS sample ≥Table F-3	Per Engineering Calculation RA-0059, the specified Table F-3 dose rates are assumed to result from radioactive iodines (I-131 thru I-135) in RCS in concentrations corresponding to the loss of 5% of gap radioactivity of the core.

Table 4 – NAPS Comparison Matrix

Category F: Fission Product Barrier Degradation

		FC Loss C.5	5. Sample line dose rate threshold ≥Table F-4	Per Engineering Calculation RA-0079, the specified Table F-4 dose rates are assumed to result from radioactive iodines (I-131 thru I-135) in RCS in concentrations corresponding to the loss of 5% of gap radioactivity of the core.
		FC Loss C.6	With letdown in service, Reactor Coolant Letdown Radiation Monitor 1(2)-CH-RI-128(228) > 7.5E+04 mrem/hr	Per Engineering Calculation PA-0234, Rev. 1, the threshold value is indicative of more than 300 µCi/cc DEI-131. A monitor reading in excess of the threshold value (7.5E+04 mrem/hr, equivalent to 300 µCi/cc) indicates a loss of the fuel clad barrier.
FC Loss 4	**CNMT Integrity or Bypass** Not Applicable	N/A	N/A	N/A
FC Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific Fuel Clad Loss indication has been identified for NAPS.
FC Loss 6	**ED Judgment** A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	FC Loss E.7	5. **Any** condition in the opinion of the SEM that indicates loss of the Fuel Clad barrier	None
FC P-Loss 1	**RCS or SG Tube Leakage** A. RCS/reactor vessel level less than (site-specific level)	FC Pot. Loss A.1	N/A	See FC Pot Loss B.1. The RCS level threshold is implemented as CSFST Core Cooling Orange Path conditions met.
FC P-Loss 2	**Inadequate Heat Removal** A. Core exit thermocouple readings greater than (site-	FC Pot. Loss B.1	2. Core Cooling-**ORANGE** Path conditions met	Consistent with the generic developers note options CSFST Core Cooling Orange Path is used in lieu of CET temperatures.

Table 4 – NAPS Comparison Matrix				
Category F: Fission Product Barrier Degradation				
	specific temperature value) **OR** B. Inadequate RCS heat removal capability via steam generators as indicated by (site-specific indications).	FC Pot. Loss B.2	3. Heat Sink-RED Path conditions met **AND** Heat sink is required	Consistent with the generic developers note options CSFST Heat Sink Red Path is used. The phrase "and heat sink required" was added to preclude the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP.
FC P-Loss 3	**RCS Activity/CMNT Rad** Not Applicable	N/A	N/A	N/A
FC P-Loss 4	**CNMT Integrity or Bypass** Not Applicable	N/A	N/A	N/A
FC P-Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific Fuel Clad Potential Loss indication has been identified for NAPS.
FC P-Loss 6	**Emergency Director Judgment** A. Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	FC Pot. Loss E.3	4. **Any** condition in the opinion of the SEM that indicates potential loss of the Fuel Clad barrier.	None

Table 4 – NAPS Comparison Matrix

Category F: Fission Product Barrier Degradation

PWR RCS Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI IC Wording	NAPS FPB #(s)	NAPS FPB Wording	Difference Justification
RCS Loss 1	**RCS or SG Tube Leakage** A. An automatic or manual ECCS (SI) actuation is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube RUPTURE.	RCS Loss A.1	1. An automatic or manual Safety Injection (SI) actuation required by **EITHER**: • UNISOLABLE RCS leakage • SG tube RUPTURE	None
RCS Loss 2	**Inadequate Heat Removal** Not Applicable	N/A	N/A	N/A
RCS Loss 3	**RCS Activity/CMNT Rad** A. Containment radiation monitor reading greater than (site-specific value).	RCS Loss C.2	2. CTMT high range radiation monitor RM-RMS-165/166(265/266) reading > Table F-2 column RCS Loss	RM-RMS-165/166(265/266) are the containment high range area radiation monitors. A reading > 5 R/hr (minimum practical reading) on RM-RMS-165/166(265/266) is indicative of a breach in the RCS barrier.
RCS Loss 4	**CNMT Integrity or Bypass** Not Applicable	N/A	N/A	N/A

Table 4 – NAPS Comparison Matrix

Category F: Fission Product Barrier Degradation

RCS Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific RCS Loss indication has been identified for NAPS.
RCS Loss 6	**Emergency Director Judgment** A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	RCS Loss E.3	3. **Any** condition in the opinion of the SEM that indicates loss of the RCS barrier	None
RCS P-Loss 1	**RCS or SG Tube Leakage** A. Operation of a standby charging (makeup) pump is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube leakage. **OR** B. RCS cooldown rate greater than (site-specific pressurized thermal shock criteria/limits defined by site-specific indications).	RCS Pot. Loss A.1	1. UNISOLABLE RCS or SG tube leakage > 150 gpm	NAPS has implemented the alternative RCS potential loss threshold provided in the generic guidance developer notes. Starting of a standby charging pump is not representative of RCS leak size relative to charging pump capacity. Nominal charging pump capacity is 150 gpm.
		RCS Pot. Loss A.2	2. Integrity-RED Path conditions met	Consistent with the generic developers note options CSFST Integrity Red Path is used.

Table 4 – NAPS Comparison Matrix

Category F: Fission Product Barrier Degradation

RCS P-Loss 2	**Inadequate Heat Removal** A. Inadequate RCS heat removal capability via steam generators as indicated by (site-specific indications).	RCS Pot. Loss B.3	3. Heat Sink-RED Path conditions met **AND** Heat sink is required	Consistent with the generic developers note options CSFST Heat Sink Red Path is used. The phrase "and heat sink required" was added to preclude the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP.
RCS P-Loss 3	**RCS Activity/CMNT Rad** Not Applicable	N/A	N/A	N/A
RCS P-Loss 4	**CNMT Integrity or Bypass** Not Applicable	N/A	N/A	N/A
RCS P-Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific RCS Potential Loss indication has been identified for NAPS.
RCS P-Loss 6	**Emergency Director Judgment** A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	RCS Pot. Loss E.4	4. **Any** condition in the opinion of the SEM that indicates potential loss of the RCS barrier	None

Table 4 – NAPS Comparison Matrix				
Category F: Fission Product Barrier Degradation				
PWR Containment Fission Product Barrier Degradation Thresholds				
NEI FPB#	NEI IC Wording	NAPS FPB #(s)	NAPS FPB Wording	Difference Justification
CNMT Loss 1	**RCS or SG Tube Leakage** A. A leaking or RUPTURED SG is FAULTED outside of containment.	CTMT Loss A.1	1. A leaking or RUPTURED SG is FAULTED outside of CTMT	None
CNMT Loss 2	**Inadequate Heat Removal** Not Applicable	N/A	N/A	N/A
CNMT Loss 3	**RCS Activity/CMNT Rad** Not applicable	N/A	N/A	N/A
CNMT Loss 4	**CNMT Integrity or Bypass** A. Containment isolation is required **AND** **EITHER** of the following: 1. Containment integrity has been lost based on	CTMT Loss D.2	2. CTMT isolation (Phase A or B) is required **AND EITHER:** • CTMT integrity has been lost based on SEM judgment • UNISOLABLE pathway from CTMT atmosphere to the environment exists	Added the word "atmosphere" to the second bulleted threshold to reinforce the generic bases that the intent is an unisolable pathway from the containment atmosphere, not RCS. RCS leakage outside containment is addressed under CTMT Loss D.3 below. Containment isolation actuation is initiated by either the Phase A or B Containment Isolation.

Table 4 – NAPS Comparison Matrix

Category F: Fission Product Barrier Degradation

	Emergency Director judgment. **OR** 2. UNISOLABLE pathway from the containment to the environment exists. **OR** B. Indications of RCS leakage outside of containment.	CTMT Loss D.3	3. Indications of UNISOLABLE RCS leakage outside of CTMT	Added the defined term "UNISOLABLE" consistent with RCS leakage thresholds to preclude transitory classifications from isolable RCS leak pathways.
CNMT Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific containment Loss indication has been identified for NAPS.
CNMT Loss 6	**Emergency Director Judgment** **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	CTMT Loss E.4	4. **Any** condition in the opinion of the SEM that indicates loss of the CTMT barrier	None
CNMT P-Loss 1	**RCS or SG Tube Leakage** Not Applicable	N/A	N/A	N/A
CNMT P-Loss 2	**Inadequate Heat Removal** A. 1. (Site-specific criteria for entry into core cooling restoration procedure) **AND** 2. Restoration procedure not effective within 15 minutes.	CTMT Pot. Loss B.1	1. Core Cooling-RED Path conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)	Consistent with the generic developers note options CSFST Core Cooling Red Path is used.

Table 4 – NAPS Comparison Matrix

Category F: Fission Product Barrier Degradation

CNMT P-Loss 3	**RCS Activity/CMNT Rad** A. Containment radiation monitor reading greater than (site-specific value).	CTMT Pot. Loss C.2	2. CTMT high range radiation monitor RM-RMS-165/166(265/266) reading > Table F-2 column CTMT Potential Loss	RM-RMS-165/166(265/266) are the containment high range area radiation monitors. The radiation monitor readings specified in Table F-2 column CTMT Potential Loss correspond to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed.
CNMT P-Loss 4	**CNMT Integrity or Bypass** A. Containment pressure greater than (site-specific value) **OR** B. Explosive mixture exists inside containment **OR** C. 1. Containment pressure greater than (site-specific pressure setpoint) **AND** 2. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer.	CTMT Pot. Loss D.4	4. Containment RED Path conditions met	Consistent with the generic developers note options CSFST Containment Red Path is used. CSFST Containment RED Path conditions are met if containment pressure exceed its design pressure. If containment pressure exceeds the design pressure of 60 psia, there exists a potential to lose the containment barrier.
		CTMT Pot. Loss D.5	5. CTMT hydrogen concentration ≥4%	A containment hydrogen concentration of 4% conservatively represents the lowest threshold for flammability in the presence of oxygen.
		CTMT Pot. Loss D.6	6. CTMT pressure > 28 psia with < one full train of CTMT depressurization equipment (Note 11) operating per design for ≥15 min. (Note 1)	The containment pressure setpoint (28 psia) is the pressure at which the containment depressurization equipment should actuate and begin performing its function. Added Note 1 consistent with other thresholds with a timing component. Added Note 11 to define what constitutes a full train of containment heat removal systems.

Table 4 – NAPS Comparison Matrix

Category F: Fission Product Barrier Degradation

CNMT P-Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific containment Potential Loss indication has been identified for NAPS.
CNMT P-Loss 6	**Emergency Director Judgment** A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.	CTMT Pot. Loss E.7	6. **Any** condition in the opinion of the SEM that indicates potential loss of the CTMT barrier	None

Table F-2 CTMT High Range Radiation Monitor Barrier Thresholds RM-RMS-165/166(265/266)

Time > Shutdown (hrs)	Fuel Clad Loss (R/hr)	RCS Loss (R/hr)	CTMT Potential Loss (R/hr)
≤2	125	5	500
> 2 – ≤4	85	5	340
> 4 – ≤6	45	5	180
> 8 – ≤14	20	5	80
> 14	10	5	40

Table F-3 FC Loss Coolant Activity Dose Rates

Time > Shutdown (hrs)	mR/hr/ml
≤2	15
> 2 – ≤8	8
> 8	3

Table F-4 FC Loss RCS Sample Line Dose Rates

Time > Shutdown (hrs)	R/hr
≤2	4
> 2 – ≤8	2
> 8	1

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
HU1	Confirmed SECURITY CONDITION or threat MODE: All	HU1	Confirmed SECURITY CONDITION or threat. MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the (site-specific security shift supervision).	HU1.1	A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by NAPS 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	Example EALs #1, 2 and 3 have been combined into a single EAL for ease of presentation and use. The "NAPS Security Shift Supervisor" is the site-specific "security shift supervision."
2	Notification of a credible security threat directed at the site.			
3	A validated notification from the NRC providing information of an aircraft threat.			

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
HU2	Seismic event greater than OBE levels MODE: All	HU2	Seismic event greater than OBE levels MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Seismic event greater than Operating Basis Earthquake (OBE) as indicated by: (site-specific indication that a seismic event met or exceeded OBE limits)	HU2.1	Seismic event > OBE (0.06g horizontal or 0.04g vertical) as indicated by "OBE EXCEEDED" indicator illuminated on the SYSCOM Network Control Center (NCC)	The "OBE EXCEEDED" indicator illuminates on the SYSCOM Network Control Center (NCC) if site OBE ground acceleration is exceeded. Ground motion acceleration of 0.06g horizontal or 0.04g vertical is the Operating Basis Earthquake for NAPS.

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
HU3	Hazardous event. MODE: All	HU3	Hazardous event MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	A tornado strike within the PROTECTED AREA.	HU3.1	A tornado strike within the PLANT PROTECTED AREA	Added the word "…PLANT…" to distinguish from the ISFSI Protected Area.
2	Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.	HU3.2	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	Changed the word "needed" to "required by Technical Specification". Plant Technical Specifications specify the needed safety systems for the current operating mode.
3	Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).	HU3.3	Movement of personnel within the PLANT PROTECTED AREA is IMPEDED due to an event external to the PLANT PROTECTED AREA involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)	Added the word "…PLANT…" to distinguish from the ISFSI Protected Area. Replaced the phrase "…due to an offsite event…" to "…due to an event external to the PLANT PROTECTED AREA…" The impact of a hazardous material originating from offsite (outside the OCA) would be the same as one originating from onsite but outside the Plant Protected Area.

Table 4 – NAPS Comparison Matrix				
Category H: Hazards and Other Conditions Affecting Plant Safety				
4	A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.	HU3.4	A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)	Added reference to Note 7.
5	(Site-specific list of natural or technological hazard events)	N/A	N/A	No other site-specific hazard has been identified for NAPS.
Note	EAL #3 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.	N/A	Note 7: This EAL does **not** apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.	This note, designated Note #7, is intended to apply to generic example EAL #4, not #3 as specified in the generic guidance.

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
HU4	FIRE potentially degrading the level of safety of the plant. MODE: All	HU4	FIRE potentially degrading the level of safety of the plant MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. A FIRE is NOT extinguished within 15-minutes of **ANY** of the following FIRE detection indications: • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms or indications • Field verification of a single fire alarm **AND** b. The FIRE is located within **ANY** of the following plant rooms or areas: (site-specific list of plant rooms or areas)	HU4.1	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	Table H-1 provides a list of site-specific fire areas.

Table 4 – NAPS Comparison Matrix				
Category H: Hazards and Other Conditions Affecting Plant Safety				
2	a. Receipt of a single fire alarm (i.e., no other indications of a FIRE). **AND** b. The FIRE is located within ANY of the following plant rooms or areas: (site-specific list of plant rooms or areas) **AND** c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.	HU4.2	Receipt of a single fire alarm (i.e., **no** other indications of a FIRE) **AND** The fire alarm is indicating a FIRE within **any** Table H-1 area **AND** The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1)	Table H-1 provides a list of site-specific fire areas. With regard to Reactor Containment fire alarms, there is constant air movement in the enclosed containment due to the operation of the containment ventilation system. The operating cooling units are drawing air to the units past the smoke detectors. It can be reasonably expected that a fire that burns for 15 minutes would produce sufficient products of combustion to cause fire detectors in multiple zones to alarm. Therefore a single containment fire alarm is not considered VALID. Added Note 13 to clarify validation of a single fire zone alarm in the Reactor Containment.
3	A FIRE within the plant *or ISFSI [for plants with an ISFSI outside the plant Protected Area]* PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.	HU4.3	A FIRE within the PLANT PROTECTED AREA or ISFSI Protected Area **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)	NAPS has an ISFSI located outside the NAPS plant Protected Area. Added the word "...PLANT..." to distinguish from the ISFSI Protected Area.
4	A FIRE within the plant *or ISFSI [for plants with an ISFSI outside the plant Protected Area]* PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.	HU4.4	A FIRE within the PLANT PROTECTED AREA or ISFSI Protected Area that requires an offsite fire department to assist with extinguishment	NAPS has an ISFSI located outside the NAPS plant Protected Area. Added the word "...PLANT..." to distinguish from the ISFSI Protected Area. Reworded example EAL #4 to better reflect the bases intent that the classification is based on a fire that requires an offsite fire department to assist with fire extinguishment.

Table 4 – NAPS Comparison Matrix				
Category H: Hazards and Other Conditions Affecting Plant Safety				
Note	**Note:** The Emergency Director should declare the Unusual Event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.
Note	N/A	N/A	Note 13: A Reactor Containment fire alarm is considered VALID upon receipt of multiple (more than one) fire zone alarms.	See justification above.

Table H-1 NAPS Fire Areas
• Cable Vaults & Tunnels • Emergency Switchgear Rooms • Emergency Diesel Generator Rooms • Reactor Containment • Quench Spray Pump Houses • Safeguards Area • Main Steam Valve House • Cable Spreading Rooms • Control Room • CR Chiller Rooms • Auxiliary / Fuel / Decontamination Buildings • Fuel Oil Pump House Room A or B • Service Water Pump House and Valve House • Intake Structure Control House • Auxiliary Service Water Pump House • Auxiliary Feedwater Pump House • Turbine Building

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
HU7	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO)UE MODE: All	HU7	Other conditions existing that in the judgment of the SEM warrant declaration of a NOUE MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.	HU7.1	Other conditions exist which in the judgment of the SEM 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.	None

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
HA1	HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes. MODE: All	HA1	HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the (site-specific security shift supervision).	HA1.1	A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by NAPS Security Shift Supervisor **<u>OR</u>** A validated notification from NRC of an aircraft attack threat within 30 min. of the site	Example EALs #1 and #2 have been combined into a single EAL for ease of use. The "NAPS Security Shift Supervisor" is the site-specific "security shift supervision."
2	A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.			

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
HA5	Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown. MODE: All	HA5	Gaseous release **IMPEDING** access to equipment necessary for normal plant operations, cooldown or shutdown MODE: 1 – Power Operations, 2 – Startup, 3 – Hot Shutdown, 4 – Hot Shutdown	Limited mode applicability to the modes specified in Table H-2.

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. Release of a toxic, corrosive, asphyxiant or flammable gas into any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode applicability identified) **AND** b. Entry into the room or area is prohibited or impeded.	HA5.1	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)	The site-specific list of plant rooms or areas with entry-related mode applicability are tabularized in Table H-2.
Note	**Note:** If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency	N/A	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	None

	classification is warranted.		is warranted.	

Table H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode**
Aux. Building El 274'	1, 2, 3, 4
Instrument Rack Rooms	4
Cable Vault & Tunnels	

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
HA6	Control Room evacuation resulting in transfer of plant control to alternate locations. MODE: All	HA6	Control Room evacuation resulting in transfer of plant control to alternate locations. MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations).	HA6.1	An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel	Auxiliary Shutdown Panel is the site-specific remote shutdown panels and local control stations.

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
HA7	Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert. MODE: All	HA7	Other conditions exist that in the judgment of the SEM warrant declaration of an Alert MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	HA7.1	Other conditions exist which, in the judgment of the SEM, 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.	None

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
HS1	HOSTILE ACTION within the PROTECTED AREA MODE: All	HS1	HOSTILE ACTION within the PLANT PROTECTED AREA MODE: All	Added the word "…PLANT…" to distinguish from the ISFSI Protected Area.

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision).	HS1.1	A HOSTILE ACTION is occurring or has occurred within the PLANT PROTECTED AREA as reported by NAPS Security Shift Supervisor	The "NAPS Security Shift Supervisor" is the site-specific "security shift supervision." Added the word "…PLANT…" to distinguish from the ISFSI Protected Area.

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
HS6	Inability to control a key safety function from outside the Control Room. MODE: All	HS6	Inability to control a key safety function from outside the Control Room MODE: 1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling	Deleted defueled mode applicability. Control of the cited safety functions are not critical for a defueled reactor as there is no energy source in the RPV or RCS. **This revised mode applicability is a deviation from the NEI 99-01 Revision 6 HS6 generic guidance but is deemed acceptable consistent with endorsed NRC EP FAQ 2015-014.**

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations). **AND** b. Control of **ANY** of the following key safety functions is not reestablished within (site-specific number of minutes). • Reactivity control • Core cooling [*PWR*] / RPV water level [*BWR*] • RCS heat removal	HS6.1	An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel **AND** Control of **any** of the following key safety functions is **not** re-established within 15 min. of the last licensed operator leaving the Control Room (Note 1): • Reactivity (Modes 1, 2 and 3 **only**) • Core Cooling • RCS heat removal	The Auxiliary Shutdown Panel is the site-specific remote shutdown panels and local control stations. Added the words "...of the last licensed operator leaving the Control Room" to provide criteria for when the 15 minutes control clock begins. The Mode applicability for the reactivity control safety function has been limited to Modes 1, 2, and 3. In Modes 4, 5 and 6, adequate shutdown margin exists under all conditions. **This revised mode applicability is a deviation from the NEI 99-01 Revision 6 HS6 generic guidance but is deemed acceptable consistent with endorsed NRC EP FAQ 2015-014.**

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
HS7	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency. MODE: All	HS7	Other conditions existing that in the judgment of the SEM warrant declaration of a Site Area Emergency MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	HS7.1	Other conditions exist which in the judgment of the SEM 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.	None

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
HG1	HOSTILE ACTION resulting in loss of physical control of the facility. MODE: All	N/A	N/A	IC HG1 and associated example EAL are not implemented in the NAPS scheme. There are several other ICs that are redundant with this IC, and are better suited to ensure timely and effective emergency declarations. In addition, the development of new spent fuel pool level EALs, as a result of NRC Order EA-12-051, clarified the intended emergency classification level for spent fuel pool level events. **This exclusion of the generic HG1 guidance is a deviation from the NEI 99-01, Revision 6 generic guidance but is deemed acceptable consistent with endorsed NRC EP FAQ 2015-013.**

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision). **AND** b. EITHER of the following has occurred: 1. **ANY** of the following safety	N/A	N/A	IC HG1 and associated example EAL is not implemented in the NAPS scheme. There are several other ICs that are redundant with this IC, and are better suited to ensure timely and effective emergency declarations. In addition, the development of new spent fuel pool level EALs, as a result of NRC Order EA-12-051, clarified the intended emergency classification level for spent fuel pool level events. This deviation is justified because:

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

	functions cannot be controlled or maintained. • Reactivity control • Core cooling [*PWR*]/RPV water level [*BWR*] • RCS heat removal **OR** 2. Damage to spent fuel has occurred or is IMMINENT.			1. Hostile Action in the Protected Area is bounded by ICs HS1 and HS7. Hostile Action resulting in a loss of physical control is bound by EAL HG7, as well as any event that may lead to radiological releases to the public in excess of Environmental Protection Agency (EPA) Protective Action Guides (PAGs). a. If, for whatever reason, the Control Room must be evacuated, and control of safety functions (e.g., reactivity control, core cooling, and RCS heat removal) cannot be reestablished, then IC HS6 would apply, as well as IC HS7 if desired by the EAL decision-maker. b. Also, as stated above, any event (including Hostile Action) that could reasonably be expected to have a release exceeding EPA PAGs would be bound by IC HG7. c. From a Hostile Action perspective, ICs HS1, HS7 and HG7 are appropriate, and therefore, make this part of HG1 redundant and unnecessary. d. From a loss of physical control perspective, ICs HS6, HS7 and HG7 are appropriate, and therefore, make this part of HG1 redundant and unnecessary. 2. Any event which causes a loss of spent fuel pool level will be bounded by ICs AA2, AS2 and AG2, regardless of whether it was based upon a Hostile Action or not, thus making this part of HG1 redundant and unnecessary. a. An event that leads to a radiological release will be bounded by ICs AU1, AA1, AS1 and AG1. Events that lead to radiological releases in excess of EPA PAGs will be bounded by EALs AG1 and HG7, thus making this part of HG1 redundant and unnecessary. ICs AA2, AS2, AG2, AS1, AG1, HS1, HS6, HS7 and HG7

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

				have been implemented consistent with NEI 99-01 Revision 6 and thus HG1 is adequately bounded as described above. **This exclusion of the generic HG1 guidance is a deviation from the NEI 99-01, Revision 6 generic guidance but is deemed acceptable consistent with endorsed NRC EP FAQ 2015-013.**

Table 4 – NAPS Comparison Matrix

Category H: Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
HG7	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency MODE: All	HG7	Other conditions exist which in the judgment of the SEM warrant declaration of a General Emergency MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	HG7.1	Other conditions exist which in the judgment of the SEM 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.	None

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SU1	Loss of all offsite AC power capability to emergency buses for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	MU1	Loss of **all** offsite AC power capability to emergency buses for 15 minutes or longer MODE: 1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Loss of **ALL** offsite AC power capability to (site-specific emergency buses) for 15 minutes or longer.	MU1.1	Loss of **all** offsite AC power capability, Table M-1, to Unit 1(2) 4160V emergency buses H and J for ≥15 min. (Note 1)	4160V emergency buses H and J are the site-specific emergency buses. Table M-1 lists credited offsite 4160V emergency bus AC power sources.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table M-1 AC Power Sources
Offsite: Unit 1 • Transfer Bus D • Transfer Bus F • Station Bus 1B • Station Bus 2B Unit 2 • Transfer Bus E • Transfer Bus F • Station Bus 2C • Station Bus 1A **Onsite:** • 1(2)H EDG • 1(2)J EDG • AAC (SBO) Diesel Generator (if already aligned)

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SU2	UNPLANNED loss of Control Room indications for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	MU3	UNPLANNED loss of Control Room indications for 15 minutes or longer. MODE: 1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.	MU3.1	An UNPLANNED event results in the inability to monitor one or more Table M-2 parameters from within the Control Room for ≥15 min. (Note 1)	The site-specific Safety System Parameter list is tabulated in Table M-2.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

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

Table M-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core exit TC temperature • Level in at least one SG • Auxiliary feedwater flow to at least one SG

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SU3	Reactor coolant activity greater than Technical Specification allowable limits. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	MU4	Reactor coolant activity greater than Technical Specification allowable limits MODE: 1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	(Site-specific radiation monitor) reading greater than (site-specific value).	MU4.1	With letdown in service, Reactor Coolant Letdown Radiation Monitor 1(2)CH-RI-128(228) > 1.50E+04 mrem/hr	Per Engineering Calculation PA-0234, Rev. 1, the threshold value is indicative of more than 60 µCi/cc DEI-131 accident mix after 1 hour of decay. A monitor reading in excess of the threshold value 1.50E+04 mrem/hr (equivalent to 60 µCi/cc) indicates a challenge to the Technical Specification allowable limits for fuel clad degradation.
		MU4.2	Dose rate at 1 ft. from an unpressurized RCS sample ≥Table M-4	Per Engineering Calculation RA-0059, dose rate is assumed to result from radioactive iodines (I-131 thru I-135) in RCS in concentrations corresponding to 60 µCi/gm DEI-131. This value corresponds to the Technical Specification coolant activity limit for iodine spike at full power operations. The values contained in Table M-4 (Tech. Spec. Coolant Activity Dose Rates) represent expected one foot dose rates per ml of sample based on time since reactor shutdown to the time when the sample is taken.
2	Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications.	MU4.3	Sample analysis indicates that a reactor coolant activity value is > an allowable limit specified in Technical Specification 3.4.16	NAPS Technical Specification 3.4.16, RCS Specific Activity, provides the Technical Specification allowable coolant activity limits.

Table M-4 Tech. Spec. Coolant Activity Dose Rates	
Time > Shutdown (hrs)	**mR/hr/ml**
≤2	0.70
> 2 – ≤8	0.50
> 8	0.30

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SU4	RCS leakage for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	MU5	RCS leakage for 15 minutes or longer MODE: 1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	RCS unidentified or pressure boundary leakage greater than (site-specific value) for 15 minutes or longer.	MU5.1	RCS unidentified or pressure boundary leakage > 10 gpm for ≥15 min. **OR** RCS identified leakage > 25 gpm for ≥15 min. **OR** Leakage from the RCS to a location outside containment > 25 gpm for ≥15 min. (Note 1)	Example EALs #1, 2 and 3 have been combined into a single EAL for usability.
2	RCS identified leakage greater than (site-specific value) for 15 minutes or longer.			
3	Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer.			
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SU5	Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor. MODE: Power Operation	MU6	Automatic or manual trip fails to shut down the reactor MODE: 1 - Power Operation	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. An automatic (trip [PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor.	MU6.1	An automatic trip did **not** shut down the reactor as indicated by reactor power ≥5% after **any** RPS setpoint is exceeded **AND** A subsequent automatic trip **OR** manual trip (trip switches or manual turbine trip) are successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)	As specified in the generic developers guidance "Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level)." Consistent with the NAPS CSFST Subcriticality Red Path criteria, a successful shutdown is defined by reactor power < 5%. Added the words "... after **any** RPS setpoint is exceeded" to clarify that it is a failure of the automatic trip when a valid scram signal has been exceeded. The reactor trip switches and manually tripping the main turbine are the means of initiating a manual trip from the reactor control consoles.

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

2	a. A manual trip ([PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. **EITHER** of the following: 1. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor. **OR** 2 A subsequent automatic (trip [PWR] / scram [BWR]) is successful in shutting down the reactor.	MU6.2	A manual trip did **not** shut down the reactor as indicated by reactor power ≥5% **AND** A subsequent manual trip (trip switches or manual turbine trip) **OR** automatic trip is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)	As specified in the generic developers guidance "Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level)." Consistent with the NAPS CSFST Subcriticality Red Path criteria, a successful shutdown is defined by reactor power < 5%. The reactor trip switches and manually tripping the main turbine are the means of initiating a manual trip from the reactor control consoles.
Notes	**Note:** A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.	N/A	Note 8: A manual action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.	None

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SU6	Loss of all onsite or offsite communications capabilities. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	MU7	Loss of **all** onsite or offsite communications capabilities. MODE: 1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Loss of **ALL** of the following onsite communication methods: (site-specific list of communications methods)	MU7.1	Loss of **all** Table M-5 onsite communication methods **OR** Loss of **all** Table M-5 State and local agency communication methods **OR** Loss of **all** Table M-5 NRC communication methods	Example EALs #1, 2 and 3 have been combined into a single EAL for simplification of presentation. Table M-5 provides a site-specific list of onsite, State and local agency (ORO) and NRC communications methods.
2	Loss of **ALL** of the following ORO communications methods: (site-specific list of communications methods)			
3	Loss of **ALL** of the following NRC communications methods: (site-specific list of communications methods)			

Table M-5 Communication Methods

System	Onsite	State/ Local	NRC
Radio Communications System	X		
Public Address and Intercom System	X		
Private Branch Telephone Exchange (PBX)	X	X	X
Sound Powered Telephone System	X		
Commercial Telephone System		X	X
Automatic Ring Downs (SONET Ring)		X	
Instaphone Loop		X	
Dedicated NRC Communications			X

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SU7	Failure to isolate containment or loss of containment pressure control. [*PWR*] MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	MU8	Failure to isolate containment or loss of containment pressure control MODE: 1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. Failure of containment to isolate when required by an actuation signal. **AND** b. ALL required penetrations are not closed within 15 minutes of the actuation signal.	MU8.1	**Any** penetration is **not** closed within 15 min. of a VALID Phase A or B isolation signal **OR** CTMT pressure > 28 psia with < one full train of CTMT depressurization equipment (Note 11) operating per design for ≥15 min. (Note 1)	Example EALs #1 and #2 have been combined for usability. Containment isolation actuation is initiated by either the Phase A or B Containment Isolation. Containment pressure greater than 28 psia is the pressure at which containment depressurization equipment are designed to automatically actuate.
2	a. Containment pressure greater than (site-specific pressure). **AND** b. Less than one full train of (site-specific system or equipment) is operating per			

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

	design for 15 minutes or longer.			
N/A	N/A	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	Added note 1 consistent with other EALs with a timing component.
N/A	N/A	N/A	Note 11: One full train of containment depressurization equipment consist of one Quench Spray (QS) System and one Recirculation Spray (RS) System from either train operating together	Added note 11 to clarify what constitutes a full train of containment heat removal systems.

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SA1	Loss of all but one AC power source to emergency buses for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	MA1	Loss of **all but one** AC power source to emergency buses for 15 minutes or longer MODE: 1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. AC power capability to (site-specific emergency buses) is reduced to a single power source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in a loss of all AC power to SAFETY SYSTEMS.	MA1.1	AC power capability, Table M-1, to Unit 1(2) 4160V emergency buses H and J 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	4160V emergency buses H and J are the site-specific emergency buses. Table M-1 lists credited offsite and onsite 4160V emergency bus AC power sources.
Note	The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table M-1 AC Power Sources
Offsite: Unit 1 • Transfer Bus D • Transfer Bus F • Station Bus 1B • Station Bus 2B Unit 2 • Transfer Bus E • Transfer Bus F • Station Bus 2C • Station Bus 1A **Onsite:** • 1(2)H EDG • 1(2)J EDG • AAC (SBO) Diesel Generator

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SA2	UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	MA3	UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress. MODE: 1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer. **AND** **ANY** of the following transient events in progress. • Automatic or manual runback greater than 25% thermal reactor power • Electrical load rejection greater than 25% full	MA3.1	An UNPLANNED event results in the inability to monitor one or more Table M-2 parameters from within the Control Room for ≥15 min. (Note 1) **AND** **Any** significant transient is in progress, Table M-3	The site-specific Safety System Parameter list is in Table M-2. The significant transient list has been tabularized in Table M-3 for ease of use.

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

	electrical load • Reactor scram [*BWR*] / trip [*PWR*] • ECCS (SI) actuation • Thermal power oscillations greater than 10% [*BWR*]			
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

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

Table M-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core exit TC temperature • Level in at least one SG • Auxiliary feedwater flow to at least one SG

Table M-3 Significant Transients
• Automatic turbine runback > 25% thermal reactor power • Electrical load rejection > 25% full electrical load • Reactor Trip • SI actuation

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SA5	Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor. MODE: Power Operation	MA6	Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are **not** successful in shutting down the reactor MODE: 1 - Power Operation	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. Manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	MA6.1	An automatic or manual trip did **not** shut down the reactor as indicated by reactor power ≥5% **AND** Subsequent automatic or manual trip actions (trip pushbuttons or manual turbine trip) are **not** successful in shutting down the reactor as indicated by reactor power ≥5% (Note 8)	As specified in the generic developers guidance "Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level)." Consistent with the NAPS CSFST Subcriticality Red Path criteria, a successful shutdown is defined by reactor power < 5%. The reactor trip pushbuttons and manually tripping the main turbine are the means of initiating a manual trip from the reactor control consoles.
Notes	**Note:** A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the	N/A	Note 8: A manual trip action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted	None

Table 4 – NAPS Comparison Matrix				
Category S: System Malfunction				
	core, and does not include manually driving in control rods or implementation of boron injection strategies.		into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies	

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SA9	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	MA9.1	Hazardous event affecting SAFETY SYSTEMS needed for the current operating mode MODE: 1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown	Revised wording from "…affecting a SAFETY SYSTEM…" to read "…affecting SAFETY SYSTEMS…" to align with changes made consistent with NRC EP FAQ 2016-002.

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. The occurrence of **ANY** of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • (site-specific hazards) • Other events with similar hazard characteristics as determined by the Shift Manager **AND** b. **EITHER** of the following: 1. Event damage has caused indications of degraded	MA9.1	The occurrence of **any** Table M-6 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 second train of the SAFETY SYSTEM needed for the current operating mode • Event damage has resulted in VISIBLE	The hazardous events have been tabularized in Table M-6. The proposed NAPS CA6.1 and MA9.1 wording is intended to ensure that an Alert should be declared only when actual or potential performance issues with SAFETY SYSTEMS have occurred as a result of a hazardous event. The occurrence of a hazardous event will result in a NOUE classification at a minimum. In order to warrant escalation to the Alert classification, the hazardous event should cause indications of degraded performance to one train of a SAFETY SYSTEM with either indications of degraded performance on the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second SAFETY SYSTEM train, such that the operability or reliability of the second train is a concern. In addition, escalation to the Alert classification should not occur if the damage from the hazardous event is limited to a SAFETY SYSTEM that was inoperable, or out of service, prior to the event occurring. As such, the proposed EALs will reduce the potential of declaring an Alert when events are in progress that do not involve an actual or potential substantial degradation of the level of safety of the plant, i.e., does not cause significant concern

Table 4 – NAPS Comparison Matrix				
Category S: System Malfunction				
	performance in at least one train of a SAFETY SYSTEM needed for the current operating mode. **OR** 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.		DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode (Notes 9, 10)	with shutting down or cooling down the plant. EALs CA6.1 and MA9.1 do not directly escalate to a Site Area Emergency or a General Emergency due to a hazardous event. The Fission Product Barrier and/or Abnormal Radiation Levels/Radiological Effluent recognition categories would provide an escalation path to a Site Area Emergency or a General Emergency. The EALs and the Basis sections have been revised to ensure potential escalations from a NOUE to an Alert, due to a hazardous event, is appropriate as the concern with these EALs is: (1) a hazardous event has occurred, (2) one SAFETY SYSTEM train is having performance issues as a result of the hazardous event, and (3) either the second SAFETY SYSTEM train is having performance issues or the VISIBLE DAMAGE is enough to be concerned that the second SAFETY SYSTEM train may have operability or reliability issues. The definition for VISIBLE DAMAGE has been revised to reflect the fact that the EALs are based upon SAFETY SYSTEM trains rather than individual components or structures. Note 9 has been added to CA6.1 and MA9.1 as it meets the intent of the EALs, is consistent with other EALs (e.g., EAL HA5.1 which was previously endorsed by the NRC), and ensures that declared emergencies are based upon unplanned events with the potential to pose a radiological risk to the public. Note 10 has been added to CA6.1 and MA9.1 to help reinforce and succinctly capture the more detailed information from the revised basis section related to when conditions would require the declaration of an Alert. CA6.1 and MA9.1 are consistent with NRC FAQ 2016-002 requiring degraded performance or visible damage to more than one safety system train caused by the specified events. **This revised wording is a deviation from the NEI 99-01**

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

				Revision 6 CA6 and SA9 generic wording and bases but is deemed acceptable consistent with endorsed NRC EP FAQ 2016-002.
N/A	N/A	N/A	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.	Added Note 9 consistent with the recommendation of NRC EP FAQ 2016-002. Added Note 10 consistent with the recommendation of NRC EP FAQ 2016-002.

Table M-6 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/SEM

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SS1	Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	MS1	Loss of **all** offsite power and **all** onsite AC power to emergency buses for 15 minutes or longer MODE: 1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Loss of **ALL** offsite and **ALL** onsite AC power to (site-specific emergency buses) for 15 minutes or longer.	MS1.1	Loss of **all** offsite and **all** onsite AC power to Unit 1(2) 4160V emergency buses H and J for ≥15 min. (Note 1)	4160V emergency buses H and J are the site-specific emergency buses.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SS5	Inability to shutdown the reactor causing a challenge to (core cooling [*PWR*] / RPV water level [*BWR*]) or RCS heat removal. MODE: Power Operation	MS6	Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal MODE: 1 - Power Operation	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. All manual actions to shutdown the reactor have been unsuccessful. **AND** c. **EITHER** of the following conditions exist: • (Site-specific indication of an inability to adequately remove heat from the core) • (Site-specific indication of an inability to adequately remove heat from the RCS)	MS6.1	An automatic or manual trip did **not** shut down the reactor as indicated by reactor power ≥5% **AND** **All** actions taken to shut down the reactor are **not** successful as indicated by reactor power ≥5% **AND EITHER:** • Core Cooling-RED PATH conditions met • Heat Sink-RED Path conditions met	As specified in the generic developers guidance "Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level)." Consistent with the NAPS CSFST Subcriticality Red Path criteria, a successful shutdown is defined by reactor power < 5%. Added the word "taken" to the second condition to emphasize the intent that it is all actions taken up to the point of either core cooling or heat sink is challenged are not successful and to not wait until all possible actions have been completed. CSFST Core Cooling-RED Path is the site-specific indication of inadequate core cooling. CSFST Heat Sink-RED Path is the site-specific indication of inadequate heat sink.

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SS8	Loss of all Vital DC power for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	MS2	Loss of **all** vital DC power for 15 minutes or longer. MODE: 1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	Indicated voltage is less than (site-specific bus voltage value) on **ALL** (site-specific Vital DC busses) for 15 minutes or longer.	MS2.1	Indicated voltage is < 105 VDC on **all** vital 125 VDC battery buses for ≥15 min. (Note 1)	105 VDC is the site-specific minimum vital 125V DC bus voltage. Vital 125 VDC battery buses are the site-specific vital DC buses credited in this EAL.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SG1	Prolonged loss of all offsite and all onsite AC power to emergency buses. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	MG1	Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses MODE: 1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. Loss of **ALL** offsite and **ALL** onsite AC power to (site-specific emergency buses). **AND** b. **EITHER** of the following: • Restoration of at least one AC emergency bus in less than (site-specific hours) is not likely. • (Site-specific indication of an inability to adequately remove heat from the core)	MG1.1	Loss of **all** offsite and **all** onsite AC power to Unit 1(2) 4160V emergency buses H and J **AND** Core Cooling-RED Path conditions met	4160V emergency buses H and J are the site-specific emergency buses. CSFST Core Cooling-RED Path is the site-specific indication of an inability to adequately remove heat from the core. The proposed NAPS MG1.1 omits the Station Blackout (SBO) coping time threshold. As proposed, the General Emergency classification would be based a loss of all onsite and offsite AC power to the emergency buses with indications of degraded core cooling. The NAPS SBO analysis and derived coping time was determined in accordance with 10CFR50.63 and Regulatory Guide 1.155. This analysis does not take credit for plant capabilities in place to mitigate the effects of an extended loss of AC power (ELAP). These capabilities were developed and implemented to meet the requirements of NRC Orders EA-12-049 and EA-12-051, and pending regulations in 10 CFR 50.155 (per SECY-16-0142). In accordance with plant EOPs [1(2)-ECA-0.0], operators will declare an ELAP within 60 min. of the loss of all AC power to the emergency buses and direct implementation of FLEX Support Guidelines,

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

				including the deployment of dedicated portable equipment and performance of DC load shedding. Even if no AC emergency bus is energized, these actions will maintain or restore core cooling, containment, and spent fuel pool cooling capabilities indefinitely. Therefore, the underlying basis for the generic EAL coping time statement, that power must be restored to an AC emergency bus within a fixed amount of time to avoid a severe challenge to one or more fission product barriers, is not valid for NAPS. **This revised wording is a deviation from the NEI 99-01, Revision 6 SG1 generic wording and bases but is deemed appropriate and acceptable.**
Note	The Emergency Director should declare the General Emergency promptly upon determining that (site-specific hours) has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table 4 – NAPS Comparison Matrix

Category S: System Malfunction

NEI IC#	NEI IC Wording	NAPS IC#(s)	NAPS IC Wording	Difference/Deviation Justification
SG8	Loss of all AC and Vital DC power sources for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	MG2	Loss of **all** emergency AC and vital DC power sources for 15 minutes or longer MODE: 1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	NAPS EAL #	NAPS EAL Wording	Difference/Deviation Justification
1	a. Loss of **ALL** offsite and **ALL** onsite AC power to (site-specific emergency buses) for 15 minutes or longer. **AND** b. Indicated voltage is less than (site-specific bus voltage value) on ALL (site-specific Vital DC busses) for 15 minutes or longer.	MG2.1	Loss of **all** offsite and **all** onsite AC power to Unit 1(2) 4160V emergency buses H and J for ≥15 min. (Note 1) **AND** Indicated voltage is < 105 VDC on **all** vital 125 VDC battery buses for ≥15 min. (Note 1)	4160V emergency buses H and J are the site-specific emergency buses. 105 VDC is the site-specific minimum vital 125V DC bus voltage. Vital 125 VDC battery buses are the site-specific vital DC buses credited in this EAL.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the NAPS EAL scheme by referencing the "time limit" specified within the EAL wording.

ATTACHMENT 2

NAPS EAL TECHNICAL BASES DOCUMENT (Marked-up)

Virginia Electric and Power Company
(Dominion Energy Virginia)
North Anna Power Station Units 1 and 2 and ISFSIs

Emergency Action Level Technical Bases Document
North Anna Power Station

(Marked-up)

Table of Contents

1.0 INTRODUCTION

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the NEI 99-01, Rev. 6, EAL Upgrade Project for North Anna Power Station (NAPS). It should be used to facilitate review of the NAPS EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EPIP-1.01, Emergency Manager Controlling Procedure, may use this document as a technical reference in support of EAL interpretation. This information may assist the Station Emergency Manager (SEM) in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training and for explaining event classifications to off-site officials.

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

Since the information in a basis document can affect emergency classification decision-making (e.g., the SEM refers to it during an event), the NRC staff expects that changes to the basis document will be evaluated in accordance with the provisions of 10 CFR 50.54(q). For Dominion Energy sites, a 10 CFR 50.54(q)(3) screening/evaluation will be performed to evaluate changes to this document.

Dominion Energy fleet procedure CM-AA-400, "10 CFR 50.59 and 10 CFR 72.48 – Changes, Tests and Experiments," provides a method to determine the impacts to licensing basis documents when changes are proposed to procedures, including changes to Abnormal Operating Procedures (AOPs) and Emergency Operating Procedures (EOPs). The 50.59/72.48 applicability review form specifically requires that the effect of a proposed procedure change on the Emergency Plan (and associated EALs) be reviewed/assessed. When impacts to the Emergency Plan are identified, a separate review in accordance to 10 CFR 50.54(q) will be performed to determine the acceptability of the proposed procedure change.

2.0 DISCUSSION

2.1 Background

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the North Anna Power Station (NAPS) Emergency Plan.

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

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

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

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

Subsequently, Rev. 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, Rev. 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 (ref. 4.1.1), NAPS conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

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

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

The primary fission product barriers are:

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

B. <u>Reactor Coolant System Barrier (RCS):</u> The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

C. <u>Containment Barrier (CTMT):</u> The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the Emergency Classification Level (ECL) from an Alert to a Site Area Emergency or a General Emergency.

2.3 Fission Product Barrier Classification Criteria

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

<u>Alert:</u>

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

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

2.4 EAL Organization

The NAPS EAL scheme includes the following features:

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

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

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

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

The EALs are pre-determined, site-specific, observable thresholds for determining whether an Initiating Condition (IC) has occurred and that an EAL threshold was met or exceeded. Thus failure to evaluate the IC and EAL together could result in an incorrect declaration.

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:	
R – Abnormal **R**ad Levels / Rad Effluent	1 – Radiological Effluent 2 – Irradiated Fuel Event 3 – Area Radiation Levels
H – **H**azards and Other Conditions Affecting Plant Safety	1 – Security 2 – Seismic Event 3 – Natural or Technological Hazard 4 – Fire 5 – Hazardous Gas 6 – Control Room Evacuation 7 – SEM Judgment
E – Independent Spent Fuel Storage Installation (ISFSI)	1 – Confinement Boundary
Hot Conditions:	
M – System **M**alfunction	1 – Loss of Emergency 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 – Containment Failure 9 – Hazardous Event Affecting Safety Systems
F – **F**ission Product Barrier Degradation	None
Cold Conditions:	
C – **C**old Shutdown / Refueling System Malfunction	1 – RCS Level 2 – Loss of Emergency AC Power 3 – RCS Temperature 4 – Loss of Vital DC Power 5 – Loss of Communications 6 – Hazardous Event Affecting Safety Systems

2.5 Technical Bases Information

EAL technical bases are provided in Attachment 1 for each EAL according to EAL group (Any, Hot, Cold), EAL category (R, C, E, F, H and M) and EAL subcategory. A summary is given at the beginning of each group, which provides a brief description of the category.

For each EAL, the following information is provided:

<u>Category Letter & Title</u>

<u>Subcategory Number & Title</u>

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

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

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

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

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

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

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

General Emergency (G), Site Area Emergency (S), Alert (A) or NOUE (U).

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

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

<u>Mode Applicability</u>

One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown, 5 - Cold Shutdown, 6 - Refueling, D - Defueled, All - All mode (See Section 2.6 for operating mode definitions).

<u>Notes (as applicable)</u>

Definitions:

If the EAL wording contains a defined term, the definition of the term is included in this section. These definitions can also be found in Section 5.1.

Basis:

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

Reference(s):

Source documentation from which the EAL is derived.

2.6 Operational Mode Applicability

Technical Specifications, definition 1.C, assigns the following reactor operating modes for Power Operation through Refueling:

1 Power Operation

K_{eff} ≥0.99 and rated thermal power > 5%

2 Startup

K_{eff} ≥0.99 and rated thermal power ≤5%

3 Hot Standby

K_{eff} < 0.99 and average reactor coolant temperature T_{avg} ≥350°F

4 Hot Shutdown

K_{eff} < 0.99 and average reactor coolant temperature 350°F > T_{avg} > 200°F with all reactor vessel head closure bolts fully tensioned

5 Cold Shutdown

K_{eff} < 0.99 and average reactor coolant temperature T_{avg} ≤200°F with all reactor vessel head closure bolts fully tensioned

6 Refueling

One or more reactor vessel head closure bolts less than fully tensioned

D Defueled

All fuel assemblies have been removed from Containment

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

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the SEM must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the EAL plus the associated Operational 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.

3.1.1 Classification Timeliness

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, "Interim Staff Guidance, Emergency Planning for Nuclear Power Plants" (ref. 4.1.8).

3.1.2 Valid Indications

All emergency classification assessments shall be based upon valid indications, reports or conditions. A valid indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy.

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

3.1.3 Imminent Conditions

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

3.1.4 Planned vs. Unplanned Events

A planned work activity that results in an expected event or condition which meets or exceeds an EAL does not warrant an emergency declaration provided that: 1) the activity proceeds as planned, and 2) the plant remains within the limits imposed by the operating license. Such activities include planned work to test, manipulate, repair, maintain or modify a system or component. In these cases, the controls associated with the planning, preparation and execution of the work will ensure that compliance is maintained with all aspects of the operating license provided that the activity proceeds and concludes as expected. Events or conditions of this type may be subject to the reporting requirements of 10CFR 50.72 (ref. 4.1.4).

3.1.5 Classification Based on Analysis

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

3.1.6 SEM Judgment

While the EALs have been developed to address a full spectrum of possible events and conditions which may warrant emergency classification, a provision for classification based on operator/management experience and judgment is still necessary. The NEI 99-01 EAL scheme provides the SEM 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 SEM will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated in the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

3.2 Classification Methodology

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

When assessing an EAL that specifies a time duration for the potentially classifiable condition, the "clock" for the EAL time duration runs concurrently with the emergency classification process "clock." For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01 (ref. 4.1.8).

3.2.1 Classification of Multiple Events and Conditions

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

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

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

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

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

3.2.2 Consideration of Mode Changes During Classification

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

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

3.2.3 Classification of Imminent Conditions

Although EALs provide specific thresholds, the SEM 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 SEM, meeting an EAL is IMMINENT, the emergency classification should be made as if the EAL has been met. While applicable to all emergency classification levels, this approach is particularly important at the higher emergency classification levels since it provides additional time for implementation of protective measures.

3.2.4 Emergency Classification Level Upgrading and Downgrading

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

As noted above, guidance concerning classification of rapidly escalating events or conditions is provided in RIS 2007-02 (ref. 4.1.2).

3.2.5 Classification of Short-Lived Events

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

3.2.6 Classification of Transient Conditions

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

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.

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

> An ATWS occurs and the high pressure ECCS systems fail to automatically start. The plant enters an inadequate core cooling condition (a potential loss of both the Fuel Clad and RCS Barriers). If an operator manually starts a high pressure ECCS system in accordance with an EOP step and clears the inadequate core cooling condition prior to an emergency declaration, then the classification should be based on the ATWS only.

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

3.2.7 After-the-Fact Discovery of an Emergency Event or Condition

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

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 (ref. 4.1.3) is applicable. Specifically, the event should be reported to the NRC in accordance with 10CFR 50.72 (ref. 4.1.4) within one hour of the discovery of the undeclared event or condition. The licensee should also notify appropriate State and local agencies in accordance with the agreed upon arrangements.

3.2.8 Retraction of an Emergency Declaration

Guidance on the retraction of an emergency declaration reported to the NRC is discussed in NUREG-1022 (ref. 4.1.3).

4.0 REFERENCES

4.1 Developmental

4.1.1 NEI 99-01, Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," (ADAMS Accession No. ML12326A805)

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

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

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

4.1.5 10 CFR 50.73, "Licensee Event Report System"

4.1.6 Technical Specifications for North Anna Units 1 and 2

4.1.7 VPAP-2103N, "Offsite Dose Calculation Manual (North Anna)"

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

4.1.9 NAPS UFSAR Section 2.1.1.3 "Boundaries for Establishing Effluent Release Limits"

4.1.10 North Anna Power Station ISFSI NRC Certificate of Compliance 1030 Amendment 1, Technical Specifications and SER

4.1.11 OU-AA-200, "Shutdown Risk Management"

4.1.12 SY-AA-101, "Security and Access Control"

4.1.13 NAPS UFSAR Section 9.1.4.3, "Fuel-Handling Structures"

4.1.14 RIS 2003-18, "Use of NEI 99-01 Methodology for Development of Emergency Action Levels" and related Supplements 1 and 2"

4.2 Implementing

4.2.1 EPIP-1.01, Emergency Manager Controlling Procedure

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

4.2.3 NAPS EAL Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

Selected terms used in Initiating Condition, EAL statements and EAL bases are set in all capital letters (e.g., ALL CAPS). These are defined terms that have specific meanings as used in this document. The definitions of these terms are provided below.

ALERT

Events are in progress, or have occurred, which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. **Any** releases are expected to be limited to small fractions of the EPA 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 NAPS ISFSI, Confinement Boundary is defined as the Sealed Surface Storage Cask (SSSC) or NUHOMS Dry Storage Canister (DSC) (ref. 4.1.10).

CONTAINMENT CLOSURE

The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions (ref. 4.1.11)~~The procedurally defined conditions or actions taken to secure containment (Primary or Secondary) and associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions~~.

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:

- Notification of Unusual Event (NOUE)
- 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.

FAULTED

The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.

FIRE

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

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 ~~a NPP~~ NAPS or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should **not** be construed to include acts of civil disobedience or felonious acts that are **not** part of a concerted attack on ~~the NPP~~NAPS. 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.

~~**Normal Levels**~~

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

NOTIFICATION of UNUSUAL EVENT

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.

OWNER CONTROLLED AREA (OCA)

The entire area contiguous to the PLANT PROTECTED AREA, owned by the Company and designated to be controlled for security reasons (ref. 4.1.12).

PLANT PROTECTED AREA

An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force (ref. 4.1.12).

PROJECTILE

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

REFUELING PATHWAY

Refueling cavity, fuel transfer canal, and spent fuel pit (SFP), but **not** including the reactor vessel, comprise the refueling pathway (ref. 4.1.13).

RUPTURED

The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

SAFETY SYSTEM

A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (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

The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment (ref. 4.1.9).

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.

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.

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

5.2 Abbreviations/Acronyms

°F .... Degrees Fahrenheit
° .... Degrees
µCi .... Micro Curie
AC .... Alternating Current
AFW .... Auxiliary Feedwater
AP .... Abnormal Procedure
ARM .... Area Radiation Monitor
ATWS .... Anticipated Transient Without Scram
CDE .... Committed Dose Equivalent
CET .... Core Exit Thermocouple
CFR .... Code of Federal Regulations
CPM .... Counts Per Minute
CR .... Control Room
CSFST .... Critical Safety Function Status Tree
CTMT .... Containment
DBA .... Design Basis Accident
DEF .... Defueled
DC .... Direct Current
DE .... Dose Equivalent
DEI-131 .... Dose Equivalent I-131
D/G .... Diesel Generator
DSC .... Dry Storage Canister
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
EDG .... Emergency Diesel Generator
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
FAA .... Federal Aviation Administration
FBI .... Federal Bureau of Investigation
FC .... Fuel Clad Barrier
FEMA .... Federal Emergency Management Agency
GE .... General Emergency
GPM .... Gallons Per Minute
Hr. .... Hour
IC .... Initiating Condition

ISFSI .... Independent Spent Fuel Storage Installation
K_{eff} .... Effective Neutron Multiplication Factor
LCO .... Limiting Condition of Operation
LOCA .... Loss of Coolant Accident
LRW .... Liquid Radwaste
LWR .... Light Water Reactor
MCB .... Main Control Board
Min. .... Minute
MPH .... Miles Per Hour
mR, mRem, mrem, mREM .... milli-Roentgen Equivalent Man
MW .... Megawatt
NEI .... Nuclear Energy Institute
NPP .... Nuclear Power Plant
NRC .... Nuclear Regulatory Commission
NSSS .... Nuclear Steam Supply System
NORAD .... North American Aerospace Defense Command
NOUE .... Notification of Unusual Event
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM .... Off-site Dose Calculation Manual
PAG .... Protective Action Guideline
PSIG .... Pounds per Square Inch Gauge
R .... Roentgen
RCS .... Reactor Coolant System
Rem, rem, REM .... Roentgen Equivalent Man
RPS .... Reactor Protection System
RVLIS .... Reactor Vessel Level Instrumentation System
SBO .... Station Blackout
SCBA .... Self-Contained Breathing Apparatus
SEM .... Station Emergency Manager
SSSC .... Sealed Surface Storage Cask
SFP .... Spent Fuel Pool (Pit)
SG .... Steam Generator
SI .... Safety Injection
SM .... Shift Manager
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
TC (T/C) .... Thermocouple
TEDE .... Total Effective Dose Equivalent

TAF .... Top of Active Fuel
TS .... Technical Specifications
TSC .... Technical Support Center
(U)FSAR .... (Updated) Final Safety Analysis Report
USGS .... United States Geological Survey

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

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

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

NAPS	NEI 99-01, Rev. 6	
EAL	IC	Example EAL
RG2.1	AG2	1
CU1.1	CU1	1
CU1.2	CU1	2
CU2.1	CU2	1
CU3.1	CU3	1
CU3.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1, 2, 3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA3.1	CA3	1, 2
CA6.1	CA6	1
CS1.1	CS1	1
CS1.2	CS1	2
CS1.3	CS1	3
CG1.1	CG1	1
CG1.2	CG1	2
EU1.1	EU1	1
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1, 2, 3
HU2.1	HU2	1

NAPS	**NEI 99-01, Rev. 6**	
EAL	**IC**	**Example EAL**
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
HG7.1	HG7	1
MU1.1	SU1	1
MU3.1	SU2	1
MU4.1	SU3	1
MU4.2	SU3	1
MU4.3	SU3	2
MU5.1	SU4	1, 2, 3
MU6.1	SU5	1

NAPS	**NEI 99-01, Rev. 6**	
EAL	**IC**	**Example EAL**
MU6.2	SU5	2
MU7.1	SU6	1, 2, 3
MU8.1	SU7	1, 2
MA1.1	SA1	1
MA3.1	SA2	1
MA6.1	SA5	1
MA9.1	SA9	1
MS1.1	SS1	1
MS2.1	SS8	1
MS6.1	SS5	1
MG1.1	SG1	1
MG2.1	SG8	1

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases

Category R – Abnormal Rad Release / Rad Effluent

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

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

At lower levels, abnormal radioactivity releases may be indicative of a failure of containment systems or precursors to more significant releases. At higher release rates, offsite radiological conditions may result which require offsite protective actions. Elevated area radiation levels in 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 required to safely operate and shutdown the plant also warrant emergency classification.

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1a – Radiological Effluent

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

EAL:

RU1.1 NOUE

Reading on SW-RM-130(230) CW Discharge Tunnel radiation monitor > 2 x the "Hi-Hi" setpoint for ≥60 min.
(Notes 1, 2, 3)

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

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

Note 3: If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

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

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

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

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

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

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

~~EAL #3 – This EAL addresses uncontrolled gaseous or liquid releases that are detected by sample analyses or environmental surveys, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.).~~

In order to optimally be able to read the "2 times the Hi-Hi alarm setpoint" threshold, the range selector switch for the monitor should be in the "wide" position. Note: This is the normal position for the switch (ref. 2).

Escalation of the emergency classification level would be via IC ~~AA1~~RA1.

Reference(s):

1. VPAP-2103N, "Offsite Dose Calculation Manual (North Anna)"
2. 1(2)-ICP-SW-RM-130(230), "Discharge Tunnel Effluent Radiation Monitor (RM-SW-( )30) Calibration"
3. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1a – Radiological Effluent

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

EAL:

RU1.2 NOUE

Sample analysis for a liquid release indicates a concentration or release rate > 2 x the allocated ODCM limits for ≥60 min.
(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

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

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

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

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

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

~~EAL #1 – This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways.~~

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

~~EAL #3 –~~This EAL addresses uncontrolled ~~gaseous or~~ liquid releases that are detected by sample analyses or environmental surveys, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in ~~river~~ lake/reservoir water systems, etc.).

Escalation of the emergency classification level would be via IC ~~AA1~~RA1.

Reference(s):

1. VPAP-2103N, "Offsite Dose Calculation Manual (North Anna)"
2. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1b – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1 mrem TEDE

EAL:

RU1.3 NOUE

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

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

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

Note 3: If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

Table R-1 Gaseous Effluent Monitor Classification Thresholds

Release Point & Monitor	GE	SAE	Alert	NOUE
Vent Stack A VG-RI-179-1 or 2	2.6E+08 µCi/sec	2.6E+07 µCi/sec	2.6E+06 µCi/sec	2.6E+05 µCi/sec
Vent Stack B VG-RI-180-1 or 2	2.0E+08 µCi/sec	2.0E+07 µCi/sec	2.0E+06 µCi/sec	2.0E+05 µCi/sec
Process Vent GW-RI-178-1 or 2	3.5E+08 µCi/sec	3.5E+07 µCi/sec	3.5E+06 µCi/sec	3.5E+05 µCi/sec
Main Steam Line MS-RI-170 (270) MS-RI-171 (271) MS-RI-172 (272)	1.3E+03 mR/hr	1.3E+02 mR/hr	1.3E+01 mR/hr	N/A
TD AFW Pump EXH MS-RI-176 (276)	6.0E+01 mR/hr	6.0E +00 mR/hr	6.0E-01 mR/hr	N/A

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

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

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

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

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

~~EAL #1 -~~ This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous ~~or liquid~~ effluent pathways (ref. 1, 2).

The basis for the NOUE values correspond to any unplanned release of gaseous effluent radioactivity to the environment that will result in greater than 1 mrem TEDE for 60 minutes or longer. This NOUE gaseous release criterion is being used consistently across all operating nuclear units at Dominion Energy. The reason this alternative criterion is required is due to the fact that for some effluent gaseous release pathways, the resulting calculated NOUE threshold following the NEI 99-01 guidance of two times the site-specific effluent release limit would result in a NOUE threshold value greater than the corresponding calculated ALERT threshold based on exceeding 10 mrem TEDE. For the other gaseous release pathways that did not show an incongruent relationship when compared to the ALERT threshold, many showed NOUE values essentially equivalent to 1 mrem TEDE when applying the guidance in NEI 99-01 of a value set at two times the site specific effluent release limit. The fact that, (1) many of the gaseous release pathway NOUE values following NEI 99-01 guidance were essentially equivalent to 1 mrem TEDE, (2) application of an alternative definition set at a value of 1 mrem TEDE results in a more limiting value for those release paths that showed incongruent comparison to the corresponding ALERT threshold, and (3) NOUE criterion set at a value ten (10) times lower than the ALERT threshold provides a logical and consistent escalation between each classification level, provides justification for the NOUE criterion of 1 mrem TEDE. This single Initiating Condition (IC) definition for gaseous releases at the NOUE level is being applied to maintain consistency across the Dominion Energy nuclear fleet and to reduce confusion and human error potential if two different (IC) definitions were applied (ref. 2). ~~EAL #2 - This EAL also addresses radioactivity releases that cause effluent radiation monitor~~

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

~~EAL #3 - This EAL addresses uncontrolled gaseous or liquid releases that are detected by sample analyses or environmental surveys, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.).~~

Classification thresholds within Table R-1 were generated using the MIDAS dose assessment code. Inputs to MIDAS use most prevalent meteorological data and expected release point parameters. An assumed one-hour decay since shutdown and a one-hour release duration are applied. Mitigating reduction mechanisms (e.g., decay, sprays, filters) input into MIDAS for each accident type determined the radiological release source term consistent with the guidance provided in NUREG-1228.

The MGPI radiation monitors for 1-GW-RI-178-1 & 2, 1-VG-RI-179-1 & 2 and 1-VG-RI-180-1 & 2 consist of a “normal” (or low) and an “accident” (or high) range device. The “normal” range radiation monitor flowpath is isolated at a predetermined value at which time the “accident” range radiation monitor is automatically aligned for operation. The “normal” range radiation monitor must be manually put back in service when flowpath activity trends down.

Due to the fact that there are no ODCM limits on steam safeties or auxiliary feedwater exhausts and the limited ability for these respective radiation monitors to detect low level radioactivity in these steam line configurations, the NOUE classification thresholds for the steam safeties and auxiliary feedwater exhaust are being labeled N/A (not applicable)

Escalation of the emergency classification level would be via IC ~~AA1~~RA1.

Reference(s):

1. VPAP-2103N, “Offsite Dose Calculation Manual (North Anna)”
2. RP 08-22, “North Anna Abnormal Rad Release Gaseous EAL Thresholds based on NEI 99-01, Rev. 6”
3. DC NA-11-01082, “Main Steam Radiation Monitor Replacement”
4. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1b – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1 mrem TEDE

EAL:

RU1.4 NOUE

Sample analysis for a gaseous release indicates a concentration or release rate > 2 x the allocated ODCM limits for ≥60 min. (Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

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

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

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

~~EAL #1 – This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways.~~

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

~~EAL #3 –~~This EAL addresses uncontrolled gaseous ~~or liquid~~ releases that are detected by sample analyses or environmental surveys~~, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.)~~.

Calculation RP 08-22 (ref. 2) demonstrates how a release rate limit based on 2 x the allocated REMODCM limit will produce essentially 1 mrem TEDE assuming most prevalent meteorological dispersion.

Most prevalent meteorology represents conditions that would most likely to exist (based on most prevalent stability class and average wind speed within that stability class). Dispersion based on most prevalent meteorology differs from that assumed in the REMODCM which uses annual average meteorology. Dispersion based on actual meteorological conditions at the time of the emergency (most prevalent) can be 10 – 20 times higher than the annual average dispersion prescribed for use in an ODCM.

Escalation of the emergency classification level would be via IC ~~AA1~~RA1.

Reference(s):

1. VPAP-2103N, "Offsite Dose Calculation Manual (North Anna)"
2. RP 08-22, "North Anna Abnormal Rad Release Gaseous EAL Thresholds based on NEI 99-01", Rev. 6
3. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.1 Alert

Reading on **any** Table R-1 effluent radiation monitor > column "ALERT" for ≥15 min. (Notes 1, 2, 3, 4)

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

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

Note 3: If the effluent flow past an effluent monitor is known to have stopped 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 RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Gaseous Effluent Monitor Classification Thresholds

Release Point & Monitor	GE	SAE	Alert	NOUE
Vent Stack A VG-RI-179-1 or 2	2.6E+08 µCi/sec	2.6E+07 µCi/sec	2.6E+06 µCi/sec	2.6E+05 µCi/sec
Vent Stack B VG-RI-180-1 or 2	2.0E+08 µCi/sec	2.0E+07 µCi/sec	2.0E+06 µCi/sec	2.0E+05 µCi/sec
Process Vent GW-RI-178-1 or 2	3.5E+08 µCi/sec	3.5E+07 µCi/sec	3.5E+06 µCi/sec	3.5E+05 µCi/sec
Main Steam Line MS-RI-170 (270) MS-RI-171 (271) MS-RI-172 (272)	1.3E+03 mR/hr	1.3E+02 mR/hr	1.3E+01 mR/hr	N/A
TD AFW Pump EXH MS-RI-176 (276)	6.0E+01 mR/hr	6.0E +00 mR/hr	6.0E-01 mR/hr	N/A

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 IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

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

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem adult thyroid CDE was established in consideration of the 1:5 ratio of the 1992 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.

Since dose assessment is based on actual meteorology whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1.

Classification thresholds within Table R-1 were generated using the MIDAS dose assessment code. Inputs to MIDAS use most prevalent meteorological data and expected release point parameters. An assumed one-hour decay since shutdown and a one-hour release duration are applied. Mitigating reduction mechanisms (e.g., decay, sprays, filters) input into MIDAS for each accident type determined the radiological release source term consistent with the guidance provided in NUREG-1228.

The MGPI radiation monitors for 1-GW-RI-178-1 & 2, 1-VG-RI-179-1 & 2 and 1-VG-RI-180-1 & 2 consist of a "normal" (or low) and an "accident" (or high) range device. The "normal" range radiation monitor flowpath is isolated at a predetermined value at which time the "accident" range radiation monitor is automatically aligned for operation. The "normal" range radiation monitor must be manually put back in service when flowpath activity trends down.

The calculated values for the Auxiliary Feedwater Pump Turbine (AFWPT) Exhaust as measured by the MGPI (i.e., Mirion Technologies) radiation monitors were slightly higher for Unit 2 than Unit 1, but within the margin of error for the radiological calculation. The Unit 1 value was used in Table R-1 for both Units 1 and 2 to simplify the table and to eliminate possibility of human error due to reading the wrong unit's value (ref. 1). Therefore, a Unit 2 event would be classified at a slightly lower value than calculated.

Escalation of the emergency classification level would be via IC ~~AS1~~RS1.

Reference(s):

1. RP 08-22, "North Anna Abnormal Rad Release Gaseous EAL Thresholds based on NEI 99-01", Rev. 6
2. DC NA-11-01082, "Main Steam Radiation Monitor Replacement"
3. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.2	**Alert**
Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem adult thyroid CDE at or beyond the SITE BOUNDARY (Note 4)	

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

Basis:

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

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

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem adult thyroid CDE was established in consideration of the 1:5 ratio of the 1992 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.~~

Since dose assessment is based on actual meteorology whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1. Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible.

Escalation of the emergency classification level would be via IC ~~AS1~~RS1.

Reference(s):

1. EPIP-4.01, "Radiological Assessment Director Controlling Procedure"
2. EPIP-4.03, "Dose Assessment Team Controlling Procedure"
3. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.3 Alert

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

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

Basis:

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

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

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 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.~~

This EAL is assessed per the ODCM (ref. 1). ODCM software can be used to produce a dose to the maximum individual.

Escalation of the emergency classification level would be via IC ~~AS1~~RS1.

Reference(s):

1. VPAP-2103N, "Offsite Dose Calculation Manual (North Anna)"
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.4 Alert

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

Basis:

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

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

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 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~~RS1.

Reference(s):

1. EPIP-4.16, “Offsite Monitoring”
2. EPIP-4.01, “Radiological Assessment Director Controlling Procedure”
3. EPIP-4.03, “Dose Assessment Team Controlling Procedure”
4. EPIP-4.34, “Field Team Radio Operator Instructions”
5. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.1 Site Area Emergency

Reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥15 min. (Notes 1, 2, 3, 4)

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

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

Note 3: If the effluent flow past an effluent monitor is known to have stopped 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 RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Gaseous Effluent Monitor Classification Thresholds

Release Point & Monitor	GE	SAE	Alert	NOUE
Vent Stack A VG-RI-179-1 or 2	2.6E+08 µCi/sec	2.6E+07 µCi/sec	2.6E+06 µCi/sec	2.6E+05 µCi/sec
Vent Stack B VG-RI-180-1 or 2	2.0E+08 µCi/sec	2.0E+07 µCi/sec	2.0E+06 µCi/sec	2.0E+05 µCi/sec
Process Vent GW-RI-178-1 or 2	3.5E+08 µCi/sec	3.5E+07 µCi/sec	3.5E+06 µCi/sec	3.5E+05 µCi/sec
Main Steam Line MS-RI-170 (270) MS-RI-171 (271) MS-RI-172 (272)	1.3E+03 mR/hr	1.3E+02 mR/hr	1.3E+01 mR/hr	N/A
TD AFW Pump EXH MS-RI-176 (276)	6.0E+01 mR/hr	6.0E +00 mR/hr	6.0E-01 mR/hr	N/A

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

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

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 EPA PAG for TEDE and thyroid CDE.

Since dose assessment is based on actual meteorology whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1.

Classification thresholds within Table R-1 were generated using the MIDAS dose assessment code. Inputs to MIDAS use most prevalent meteorological data and expected release point parameters. An assumed one-hour decay since shutdown and a one-hour release duration are applied. Mitigating reduction mechanisms (e.g., decay, sprays, filters) input into MIDAS for each accident type determined the radiological release source term consistent with the guidance provided in NUREG-1228.

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.

The MGPI radiation monitors for 1-GW-RI-178-1 & 2, 1-VG-RI-179-1 & 2 and 1-VG-RI-180-1 & 2 consist of a "normal" (or low) and an "accident" (or high) range device. The "normal" range radiation monitor flowpath is isolated at a predetermined value at which time the "accident" range radiation monitor is automatically aligned for operation. The "normal" range radiation monitor must be manually put back in service when flowpath activity trends down.

The calculated values for the Auxiliary Feedwater Pump Turbine (AFWPT) Exhaust as measured by the MGPI (i.e., Mirion Technologies) radiation monitors were slightly higher for Unit 2 than Unit 1, but within the margin of error for the radiological calculation. The Unit 1 value was used in Table R-1 for both Units 1 and 2 to simplify the table and to eliminate possibility of human error due to reading the wrong unit's value (ref. 1). Therefore, a Unit 2 event would be classified at a slightly lower value than calculated.

Escalation of the emergency classification level would be via IC ~~AG1~~RG1.

Reference(s):

1. RP 08-22, "North Anna Abnormal Rad Release Gaseous EAL Thresholds based on NEI 99-01", Rev. 6
2. DC NA-11-01082, "Main Steam Radiation Monitor Replacement"
3. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.2 Site Area Emergency

Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem adult thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

Basis:

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

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

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 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.~~

Since dose assessment is based on actual meteorology whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1. Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible.

Escalation of the emergency classification level would be via IC AG1.

Reference(s):

1. EPIP-4.01, “Radiological Assessment Director Controlling Procedure”
2. EPIP-4.03, “Dose Assessment Team Controlling Procedure”
3. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.3 Site Area Emergency

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

Basis:

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

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

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 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~~RG1.

Reference(s):

1. EPIP-4.16, "Offsite Monitoring"
2. EPIP-4.01, "Radiological Assessment Director Controlling Procedure"
3. EPIP-4.03, "Dose Assessment Team Controlling Procedure"
4. EPIP-4.34, "Field Team Radio Operator Instructions"
5. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.1 General Emergency

Reading on **any** Table R-1 effluent radiation monitor > column "GE" for ≥15 min. (Notes 1, 2, 3, 4)

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

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

Note 3: If the effluent flow past an effluent monitor is known to have stopped 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 RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Gaseous Effluent Monitor Classification Thresholds

Release Point & Monitor	GE	SAE	Alert	NOUE
Vent Stack A VG-RI-179-1 or 2	2.6E+08 µCi/sec	2.6E+07 µCi/sec	2.6E+06 µCi/sec	2.6E+05 µCi/sec
Vent Stack B VG-RI-180-1 or 2	2.0E+08 µCi/sec	2.0E+07 µCi/sec	2.0E+06 µCi/sec	2.0E+05 µCi/sec
Process Vent GW-RI-178-1 or 2	3.5E+08 µCi/sec	3.5E+07 µCi/sec	3.5E+06 µCi/sec	3.5E+05 µCi/sec
Main Steam Line MS-RI-170 (270) MS-RI-171 (271) MS-RI-172 (272)	1.3E+03 mR/hr	1.3E+02 mR/hr	1.3E+01 mR/hr	N/A
TD AFW Pump EXH MS-RI-176 (276)	6.0E+01 mR/hr	6.0E +00 mR/hr	6.0E-01 mR/hr	N/A

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

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

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 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.

Since dose assessment is based on actual meteorology whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1.

Classification thresholds within Table R-1 were generated using the MIDAS dose assessment code. Inputs to MIDAS use most prevalent meteorological data and expected release point parameters. An assumed one-hour decay since shutdown and a one-hour release duration are applied. Mitigating reduction mechanisms (e.g., decay, sprays, filters) input into MIDAS for each accident type determined the radiological release source term consistent with the guidance provided in NUREG-1228.

The MGPI radiation monitors for 1-GW-RI-178-1 & 2, 1-VG-RI-179-1 & 2 and 1-VG-RI-180-1 & 2 consist of a "normal" (or low) and an "accident" (or high) range device. The "normal" range radiation monitor flowpath is isolated at a predetermined value at which time the "accident" range radiation monitor is automatically aligned for operation. The "normal" range radiation monitor must be manually put back in service when flowpath activity trends down.

The calculated values for the Auxiliary Feedwater Pump Turbine (AFWPT) Exhaust as measured by the MGPI (i.e., Mirion Technologies) radiation monitors were slightly higher for Unit 2 than Unit 1, but within the margin of error for the radiological calculation. The Unit 1 value was used in Table R-1 for both Units 1 and 2 to simplify the table and to eliminate possibility of human error due to reading the wrong unit's value (ref. 1). Therefore, a Unit 2 event would be classified at a slightly lower value than calculated.

Reference(s):

1. RP 08-22, "North Anna Abnormal Rad Release Gaseous EAL Thresholds based on NEI 99-01", Rev. 6
2. DC NA-11-01082, "Main Steam Radiation Monitor Replacement"
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.2	**General Emergency**
Dose assessment using actual meteorology indicates doses > 1,000 mrem TEDE or 5,000 mrem adult thyroid CDE at or beyond the SITE BOUNDARY (Note 4)	

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

Basis:

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

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

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 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.~~

Since dose assessment is based on actual meteorology whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1. Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible.

Reference(s):

1. EPIP-4.01, "Radiological Assessment Director Controlling Procedure"
2. EPIP-4.03, "Dose Assessment Team Controlling Procedure"
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.3 General Emergency

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

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

(Notes 1; 2)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

Basis:

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

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

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 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. EPIP-4.16, "Offsite Monitoring"
2. EPIP-4.01, "Radiological Assessment Director Controlling Procedure"

3. EPIP-4.03, "Dose Assessment Team Controlling Procedure"
4. EPIP-4.34, "Field Team Radio Operator Instructions"
5. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: UNPLANNED loss of water level above irradiated fuel

EAL:

RU2.1 NOUE

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

- Spent Fuel Pit Lo Level (1E-C6) alarm
- Report of dropping level in refueling cavity or SFP
- Loss of SFP Cooling suction flow

AND

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

- RM-RMS-152 New Fuel Storage Area
- RM-RMS-153 Fuel Pit Bridge
- RM-RMS-162 (262) Manipulator Crane Area (Refueling Mode)
- RM-RMS-163 (263) Reactor Containment Area

Mode Applicability:

All

Definition(s):

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

REFUELING PATHWAY- Refueling cavity, fuel transfer canal, and spent fuel pit (SFP), but **not** including the reactor vessel, comprise the refueling pathway.

Basis:

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

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

The SFP level is remotely monitored by level switches FC-LS-100 (high) and 101 (low). The level switch initiates high and low level annunciators. The SFP WATER LEVEL LOW alarm (window 1E-C6) actuates if SFP level decreases to the 289 ft 4 in. el. Local level indication is provided by a ruled scale mounted on the east side of the counterfort. Normal level is indicated by the 0 mark on the scale and corresponds to 289 ft 10 in. el. or normal SFP level. Level is normally maintained between the 0 in. mark and the +3 in. mark. The low level alarm corresponds to the -6 in. mark (ref. 1, 2).

The Spent Fuel Pool (SFP) wide-range level indication system is available to monitor water level. Two (2) level instruments are installed in the SFP with indicators, 1-FC-LI-105-1, 2 & 2A provided in the Main Control Room and MCR Computer Rooms. The level instruments will provide level indication over the entire span of the SFP from the top of the fuel racks to 10 inches above the normal operating level (ref. 5).

The specified radiation monitors are those expected to see increase area radiation levels as a result of a loss of REFUELING PATHWAY inventory (ref. 4). Increasing radiation indications on these monitors in the absence of indications of decreasing REFUELING PATHWAY level are not classifiable under this EAL.

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 with ~~Recognition~~ Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC ~~AA2~~RA2.

Reference(s):

1. AR 1-E-C6, “Spent Fuel Pit Lo Level”
2. 0-AP-27, “Malfunction of Spent Fuel Pit Systems”
3. 1(2)-AP-52, “Loss of Refueling Cavity Level During Refueling
4. 0-AP-5.1, “Common Unit Radiation Monitoring System”
5. Design Change NA-13-01043, “BDB Spent Fuel Pool Level Instrumentation Installation – Units 1 & 2”
6. NEI 99-01 AU2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.1 Alert

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 NAPS ISFSI, Confinement Boundary is defined as the Sealed Surface Storage Cask (SSSC) or NUHOMS Dry Storage 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- Refueling cavity, fuel transfer canal, and spent fuel pit (SFP), but **not** including the reactor vessel, comprise the refueling pathway.

Basis:

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

~~This IC applies to~~For irradiated fuel that is licensed for dry storage, this EAL applies up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC E-~~H~~U1.

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

~~EAL #1~~

This EAL escalates from ~~AU2~~ RU2.1 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used (e.g., a boil-off curve). Classification of an event using this EAL should be based on the totality of available indications, reports and observations.

While an area radiation monitor could detect an increase in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance with ~~Recognition~~ Category C during the Cold Shutdown and Refueling modes.~~.EAL #2~~

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

~~EAL #3~~

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

~~Escalation of the emergency classification level would be via ICs AS1 or AS2 (*see AS2 Developer Notes*).~~

Reference(s):

1. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity

AND EITHER:

- VALID Hi-Hi alarm on **any** of the following radiation monitors:
  - RM-RMS-152 New Fuel Storage Area
  - RM-RMS-153 Fuel Pit Bridge
  - RM-RMS-162 (262) Manipulator Crane Area (Refueling Mode)
  - RM-RMS-163 (263) Reactor Containment Area
  - RM-RMS-159 (259) Containment Particulate
  - RM-RMS-160 (260) Containment Area Gas
- VALID Hi alarm on VG-RI-180-1 Vent Stack B Normal Range

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 NAPS ISFSI, Confinement Boundary is defined as the Sealed Surface Storage Cask (SSSC) or NUHOMS Dry Storage Canister (DSC).

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:

The specified radiation monitors are those expected to see increased area radiation levels as a result of damage to irradiated fuel (ref. 1, 2, 3, 4).

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

~~This IC applies to~~For irradiated fuel that is licensed for dry storage, this EAL applies up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC E-~~H~~U1.

~~EAL #This EAL escalates from AU2 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used (e.g., a boil-off curve). Classification of an event using this EAL should be based on the totality of available indications, reports and observations.~~

~~While an area radiation monitor could detect an increase in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.~~

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

This EAL addresses a release of radioactive material caused by mechanical damage to irradiated fuel. Damaging events may include the dropping, bumping or binding of an assembly, or dropping a heavy load onto an assembly. A rise in readings on radiation monitors should be considered in conjunction with in-plant reports or observations of a potential fuel damaging event (e.g., a fuel handling accident). ~~EAL #3Spent 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 would be based on either ~~Recognition~~ Category ~~A~~ R or C ICs. ~~Escalation of the emergency classification level would be via ICs AS1or AS2 (*see AS2 Developer Notes*).~~

Reference(s):

1. 1(2)-AP-5, “Unit 1(2) Radiation Monitoring System”
2. 0-AP-5.1, “Common Unit Radiation Monitoring System”
3. 0-AP-5.2, “MGP Radiation Monitoring System”
4. 0-AP-30, “Fuel Failure During Handling”
5. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.3 Alert

Lowering of spent fuel pool level to 10 ft. (Level 2) on 1-FC-LI-105-1, 2 or 2A Spent Fuel Pit Wide Range Level

Mode Applicability:

All

Definition(s):

None

Basis:

This ~~IC~~ EAL addresses events that have caused ~~IMMINENT or actual damage to an irradiated fuel assembly, or~~ a significant lowering of water level within the spent fuel pool ~~*(see Developer Notes)*~~. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant. ~~This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC E-HU1.~~

~~Escalation of the emergency would be based on either Recognition Category A or C ICs.~~ EAL # ~~This EAL escalates from AU2 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used (e.g., a boil-off curve). Classification of an event using this EAL should be based on the totality of available indications, reports and observations.~~

~~While an area radiation monitor could detect an increase in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.~~

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

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

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

Escalation of the emergency classification level would be via ICs ~~AS1~~ RS1 or ARS2 ~~(see AS2 Developer Notes)~~.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (1-FC-LI-105-1, 1-FC-LI-105-2 and 1-FC-LI-105-2A) capable of identifying normal level (Level 1 –EL 289 ft. 10 in.), SFP level 10 ft. above the top of the fuel racks (Level 2 –EL 274 ft. 8 in.) and SFP level at 1 foot above the top of the fuel racks (Level 3 –EL 265 ft. 8 in.) (ref. 1, 2, 3).

Level	**Plant Elevation**	**1-FC-LI-105-1, 2 or 2A Reading** (ft. above top of spent fuel racks)
1	289 ft. 10 in.	25.2 ft.
2	274 ft. 8 in.	10 ft.
3	265 ft. 8 in.	1 ft.

Reference(s):

1. ETE-CPR-2012-0012, “North Anna Units 1 & 2 – Beyond Design Basis FLEX Strategy Basis Document and Final Integration Plan”
2. DC NA-13-01043, “Beyond Design Basis Spent Fuel Pool Level Instrument Installation – North Anna Units 1 & 2”
3. 0-AP-27, “Malfunction of Spent Fuel Pit Systems”
4. 1(2)-AP-52, “Loss of Refueling Cavity Level During Refueling”
5. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RS2.1 Site Area Emergency

Lowering of spent fuel pool level to 1 ft. (Level 3) on 1-FC-LI-105-1, 2 or 2A Spent Fuel Pit Wide Range Level

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 ~~IC~~ 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~~ RG1 or ~~AG2~~RG2.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (1-FC-LI-105-1, 1-FC-LI-105-2 and 1-FC-LI-105-2A) capable of identifying normal level (Level 1 –EL 289 ft. 10 in.), SFP level 10 ft. above the top of the fuel racks (Level 2 –EL 274 ft. 8 in.) and SFP level at 1 foot above the top of the fuel racks (Level 3 –EL 265 ft. 8 in.) (ref. 1, 2, 3).

Level	Plant Elevation	1-FC-LI-105-1, 2 or 2A Reading (ft. above top of spent fuel racks)
1	289 ft. 10 in.	25.2 ft.
2	274 ft. 8 in.	10 ft.
3	265 ft. 8 in.	1 ft.

Reference(s):

1. ETE-CPR-2012-0012, "North Anna Units 1 & 2 – Beyond Design Basis FLEX Strategy Basis Document and Final Integration Plan"
2. DC NA-13-01043, "Beyond Design Basis Spent Fuel Pool Level Instrument Installation – North Anna Units 1 & 2"
3. 0-AP-27. "Malfunction of Spent Fuel Pit Systems"
4. 1(2)-AP-52, "Loss of Refueling Cavity Level During Refueling"
5. NEI 99-01 AS2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RG2.1 General Emergency

Spent fuel pool level **cannot** be restored to at least 1 ft. (Level 3) on 1-FC-LI-105-1, 2 or 2A Spent Fuel Pit Wide Range Level for ≥60 min.
(Note 1)

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

Mode Applicability:

All

Definition(s):

None

Basis:

This ~~IC~~ 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~~ EAL would likely not be met until well after another General Emergency ~~IC~~ EAL was met; however, it is included to provide classification diversity.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (1-FC-LI-105-1, 1-FC-LI-105-2 and 1-FC-LI-105-2A) capable of identifying normal level (Level 1 –EL 289 ft. 10 in.), SFP level 10 ft. above the top of the fuel racks (Level 2 –EL 274 ft. 8 in.) and SFP level at 1 foot above the top of the fuel racks (Level 3 –EL 265 ft. 8 in.) (ref. 1, 2, 3).

Level	Plant Elevation	1-FC-LI-105-1, 2 or 2A Reading (ft. above top of spent fuel racks)
1	289 ft. 10 in.	25.2 ft.
2	274 ft. 8 in.	10 ft.
3	265 ft. 8 in.	1 ft.

Reference(s):

1. ETE-CPR-2012-0012, "North Anna Units 1 & 2 – Beyond Design Basis FLEX Strategy Basis Document and Final Integration Plan"
2. DC NA-13-01043, "Beyond Design Basis Spent Fuel Pool Level Instrument Installation – North Anna Units 1 & 2"
3. 0-AP-27, "Malfunction of Spent Fuel Pit Systems"
4. 1(2)-AP-52, "Loss of Refueling Cavity Level During Refueling"
5. NEI 99-01 AG2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

RA3.1 Alert

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

- Control Room
- Central Alarm Station (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:

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The ~~Emergency Director~~SEM should consider the cause of the increased radiation levels and determine if another IC may be applicable. ~~For EAL #2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the increased radiation levels. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., installing temporary shielding, requiring use of non-routine protective equipment, requesting an extension in dose limits beyond normal administrative limits).~~

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

~~The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the elevated radiation levels). For example, the plant is in Mode 1 when the radiation increase occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.~~

~~The increased radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).~~

~~The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).~~

~~The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.~~

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, RM-RMS-157 (ref. 1). 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 that may be used to assess this EAL threshold. Therefore, this threshold is evaluated using local radiation survey for this area.

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

Reference(s):

1. 0-AP-5.1, "Common Unit Radiation Monitoring System"
2. NEI 99-01 AA3

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

RA3.2 Alert

An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 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 R-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode**
Aux. Building El 274'	1, 2, 3, 4
Instrument Rack Rooms	4
Cable Vault & Tunnels	

Mode Applicability:

1 - Power Operation, 2 – Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective 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 IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The ~~Emergency Director~~SEM should consider the cause of the increased radiation levels and determine if another IC may be applicable.

For ~~EAL #2~~ RA3.2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the increased radiation levels. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., installing temporary shielding, requiring use of non-routine protective equipment, requesting an extension in dose limits beyond normal administrative limits).

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).

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

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

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

Reference(s):

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

Category C – Cold Shutdown / Refueling System Malfunction

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

Category C EALs are directly associated with cold shutdown or refueling system safety functions. Given the variability of plant configurations (e.g., systems out-of-service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The cold shutdown and refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, CONTAINMENT CLOSURE, and fuel clad integrity for the applicable operating modes (5 - Cold Shutdown, 6 - Refueling, DEF – Defueled).

The events of this category pertain to the following subcategories:

1. <u>RCS Level</u>

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

2. <u>Loss of Emergency AC Power</u>

   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 4160V AC emergency buses.

3. <u>RCS Temperature</u>

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

4. <u>Loss of Vital DC Power</u>

   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. <u>Loss of Communications</u>

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

6. <u>Hazardous Event Affecting Safety Systems</u>

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

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory

EAL:

CU1.1 NOUE

UNPLANNED loss of reactor coolant results in RCS water level less than a required lower limit for ≥15 min. (Note 1)

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

Mode Applicability:

5 - Cold Shutdown, 6 - 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:

RCS water level less than a required lower limit is meant to be less than the lower end of the level control band being procedurally maintained for the current condition or evolution.

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

With the plant in Refueling mode, RCS water level is normally maintained at or above the reactor vessel flange (ref. 2).

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

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

This EAL ~~#1~~ recognizes that the minimum required ~~(reactor vessel/~~RCS ~~[PWR] or RPV [BWR])~~ 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.

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

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

Reference(s):

1. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
2. 1(2)-OP-4.1, "Controlling Procedure for Refueling"
3. 1(2)-AP-17, "Shutdown LOCA"
4. 1(2)-AP-11, "Loss of RHR"
5. 1(2)-AP-52, "Loss of Refueling Cavity Level During Refueling"
6. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory

EAL:

CU1.2 NOUE

RCS water level **cannot** be monitored

AND EITHER:

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

Table C-1 Sumps/Tanks
• Reactor Containment Sump • Pressurizer Relief Tank (PRT) • Primary Drain Transfer Tank (PDTT) • Component Cooling (CC) Surge Tank • Refueling Water Storage Tank (RWST)

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

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

UNPLANNED-. A parameter changes 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 IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor ~~(reactor vessel/~~RCS ~~[PWR] or RPV [BWR])~~ level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

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

~~EAL #1 recognizes that the minimum required (reactor vessel/RCS [PWR] or [BWR]) 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.~~

This EAL ~~#2~~ addresses a condition where all means to determine ~~(reactor vessel/~~RCS ~~[PWR] or [BWR])~~ level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels (Table C-1) (ref. 1, 2, 3, 4, 5). Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the ~~(reactor vessel/~~RCS ~~[PWR] or [BWR])~~.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refueling mode, normal means of RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

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

Reference(s):

1. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
2. 1(2)-OP-4.1, "Controlling Procedure for Refueling"
3. 1(2)-AP-17, "Shutdown LOCA"
4. 1(2)-AP-11, "Loss of RHR"
5. 1(2)-AP-52, "Loss of Refueling Cavity Level During Refueling"
6. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Significant Loss of RCS inventory

EAL:

CA1.1	**Alert**
RCS level < minimum required for continued RHR pump operation	

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

None

Basis:

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

For this EAL ~~#1~~, a lowering of RCS water level below ~~(site-specific level) ft~~ the specified value(s) indicates that operator actions have not been successful in restoring and maintaining ~~(reactor vessel/~~RCS ~~[PWR] or RPV [BWR])~~ 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. The classification threshold is based on the lowest RCS level that supports continued decay heat removal pump (RHR) operations per procedure (ref. 1, 2, 3, 4).

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

~~For EAL #2, the inability to monitor (reactor vessel/RCS [PWR] or RPV [BWR]) level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the (reactor vessel/RCS [PWR] or RPV [BWR]).~~

~~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 RCS ~~the (reactor vessel/RCS [PWR] or RPV [BWR]) inventory~~ water level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

Reference(s):

1. 1(2)-OP-5.4, "Draining the Reactor Coolant System"

2. 1(2)-AP-17, "Shutdown LOCA"
3. 1(2)-AP-11, "Loss of RHR"
4. 1(2)-AP-52, "Loss of Refueling Cavity Level During Refueling"
5. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Significant Loss of RCS inventory

EAL:

CA1.2 Alert

RCS water level **cannot** be monitored for ≥15 min. (Note 1)

AND EITHER

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

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

Table C-1 Sumps/Tanks
• Reactor Containment Sump
• Pressurizer Relief Tank (PRT)
• Primary Drain Transfer Tank (PDTT)
• Component Cooling (CC) Surge Tank
• Refueling Water Storage Tank (RWST)

Mode Applicability:

5 - Cold Shutdown, 6 - 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.

Basis:

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

~~For EAL #1, a lowering of water level below (site-specific level) indicates that operator actions have not been successful in restoring and maintaining (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.~~

~~Although related, EAL #1 is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Residual Heat Removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.~~

For this EAL ~~#2~~, the inability to monitor ~~(reactor vessel/~~RCS ~~[PWR] or RPV [BWR])~~ level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level (Table C-1) changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the ~~(reactor vessel/~~RCS ~~[PWR] or RPV [BWR])~~ (ref 1, 2, 3, 4, 5).

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refueling mode, normal means of RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

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

If the ~~(reactor vessel/~~RCS ~~[PWR] or RPV [BWR])~~ inventory level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

Reference(s):

1. 1(2)-OP-5.4, “Draining the Reactor Coolant System”
2. 1(2)-OP-4.1, “Controlling Procedure for Refueling”
3. 1(2)-AP-17, “Shutdown LOCA”
4. 1(2)-AP-11, “Loss of RHR”
5. 1(2)-AP-52, “Loss of Refueling Cavity Level During Refueling”
6. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.1	**Site Area Emergency**
With CONTAINMENT CLOSURE **not** established, **any** confirmed loss of inventory indication, Table C-2, with RVLIS full range < 62%	

Table C-2 Inventory Loss Confirmatory Indications
• In service Standpipe and Ultrasonic level bottomed out
• Decreasing RVLIS level trend
• RHR pump amp fluctuations

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions.

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

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

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS/reactor vessel levels of EALs ~~1.b~~CS1.1 and ~~2.b~~CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment (ref. 1).

~~In EAL 3.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows~~

~~sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.~~

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

This~~ese~~ EAL~~s~~ 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.

When Reactor Vessel water level decreases to 254.625 ft el., water level is six inches below the elevation of the bottom of the RCS hot leg penetration. When Reactor Vessel water level drops significantly below the elevation of the bottom of the RCS hot leg penetration, all sources of RCS injection have failed or are incapable of making up for the inventory loss. Six inches below the elevation of the bottom of the RCS hot leg penetration can be monitored only by RVLIS full range (62.0%). Level monitoring instruments 1-RC-LI-102 (2-RC-LI-202), 1-RC-LI-103, (2-RC-LI-203) 1-RC-LI-105 (2-RC-LI-205) and RVLIS upper range are offscale low when level is below the elevation of the centerline of the RCS loop hot leg penetration (256.333 ft el.).

Table C-2 provides a list of confirmatory indicators for RCS inventory loss. Due to the variability of accuracy and usability of RVLIS while in Cold Shutdown or Refueling Mode, the use of RVLIS for emergency classification purposes is contingent on one or more of the listed confirmatory indications.

The RVLIS full range threshold has been determined as follows (ref. 2, 3, 4):

Component	Elevation (ft)	Radius (in.)	RVLIS Full Range (%)
RCS hot leg centerline	256.333	14.5	63.0
Bottom of RCS hot leg	255.125	NA	A
6 in. below bottom of hot leg	254.625	NA	B
Top of fuel	252.807	NA	61.0

RVLIS span %/ft = 0.56721

A = 61.0% + (Bottom of RCS hot leg - Top of fuel) x RVLIS span
= 62.3%

B = 61.0% + (6 in. below bottom of hot leg - Top of fuel) x RVLIS span
= 62.0%

EAL RVLIS values have been rounded up to the nearest whole percentage point.

Escalation of the emergency classification level would be via ICs CG1 or ~~AG1~~RG1.

Reference(s):

1. OU-AA-200, "Shutdown Risk Management"
2. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
3. 1(2)-OP-4.1, "Controlling Procedure for Refueling"
4. 1(2)-AP-17, "Shutdown LOCA"
5. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.2 Site Area Emergency

With CONTAINMENT CLOSURE established, **any** confirmed loss of inventory indication, Table C-2, with RVLIS full range < 61%

Table C-2 Inventory Loss Confirmatory Indications
• In service Standpipe and Ultrasonic level bottomed out • Decreasing RVLIS level trend • RHR pump amp fluctuations

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions.

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

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

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS/reactor vessel levels of EALs ~~1.b~~CS1.1 and ~~2.b~~CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment (ref. 1).

~~In EAL 3.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows~~

~~sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.~~

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

Thi~~ese~~s EAL~~s~~ 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.

This level drop can only be remotely monitored by Reactor Vessel Level Instrumentation System (RVLIS). When Reactor Vessel water level drops below RVLIS full range setpoint of 61% (ref. 2), core uncovery is about to occur.

Table C-2 provides a list of confirmatory indicators for RCS inventory loss. Due to the variability of accuracy and usability of RVLIS while in Cold Shutdown or Refueling Mode, the use of RVLIS for emergency classification purposes is contingent on one or more of the listed confirmatory indications.

Escalation of the emergency classification level would be via ICs CG1 or ~~AG1~~RG1.

Reference(s):

1. OU-AA-200, “Shutdown Risk Management”
2. 1(2)-OP-5.4, “Draining the Reactor Coolant System”
3. 1(2)-OP-4.1, “Controlling Procedure for Refueling”
4. 1(2)-AP-17, “Shutdown LOCA”
5. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.3 Site Area Emergency

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

AND

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

- UNPLANNED increase in **any** Table C-1 sump or tank level of sufficient magnitude to indicate core uncovery
- Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery
- **Any** containment area radiation monitor reading > 3 R/hr (Refueling Mode)
- Erratic source range monitor indications

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

Table C-1 Sumps/Tanks
• Reactor Containment Sump • Pressurizer Relief Tank (PRT) • Primary Drain Transfer Tank (PDTT) • Component Cooling (CC) Surge Tank • Refueling Water Storage Tank (RWST)

Mode Applicability:

5 - Cold Shutdown, 6 - 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.

Basis:

This IC addresses a significant and prolonged loss of ~~(reactor vessel/RCS~~ RCS ~~[PWR] or RPV [BWR])~~ 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 RCS~~/reactor vessel~~ level cannot be restored, fuel damage is probable.

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

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

The inability to monitor ~~(reactor vessel/~~RCS ~~[PWR] or RPV [BWR])~~ level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels (Table C-1). Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the ~~(reactor vessel/~~RCS ~~[PWR] or RPV [BWR])~~ (ref. 1, 2, 3).

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refueling mode, normal means of RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory.

In the Refueling mode, the dose rate above the core will rise as water level in the reactor vessel lowers. The dose rate due to this core shine should result in on-scale indications of > 3 R/hr on containment area radiation monitors (ref. 4).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.

~~These~~ This EAL~~s~~ 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 ICs CG1 or ~~AG1~~RG1

Reference(s):

1. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
2. 1(2)-OP-4.1, "Controlling Procedure for Refueling"
3. 1(2)-AP-17, "Shutdown LOCA"
4. RA-0078, "Verification of Radiation Monitor Response to Core Uncovery"
5. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CG1.1 General Emergency

Any confirmed loss of inventory indication, Table C-2, with RVLIS full range < 61% for ≥30 min. (Note 1)

AND

Any Containment Challenge indication, Table C-3

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

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

Table C-2 Inventory Loss Confirmatory Indications
• In service Standpipe and Ultrasonic level bottomed out • Decreasing RVLIS level trend • RHR pump amp fluctuations

Table C-3 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • CTMT hydrogen concentration ≥4% • UNPLANNED increase in CTMT pressure

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions.

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:

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

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

Three conditions are associated with a challenge to containment's capability to serve as an effective barrier to fission product release (Table C-3):

1. With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment (ref. 1). If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.
2. The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit of 4%). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to containment integrity.

   In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. However, 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 burn (4%) (ref. 2). If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.
3. Any UNPLANNED rise in containment pressure in the Cold Shutdown or Refueling mode indicates a potential challenge of CONTAINMENT CLOSURE capability. This is due to the potential use of temporary penetration seals, water seals or other closure mechanisms used to support maintenance that are not suitable to withstand a rise in containment pressure. UNPLANNED containment pressure rise indicates CONTAINMENT CLOSURE cannot be assured and the containment cannot be relied upon as a barrier to fission product release.

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

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

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*.

This level drop can only be remotely monitored by Reactor Vessel Level Instrumentation System (RVLIS). When Reactor Vessel water level drops below RVLIS full range setpoint of 61%, core uncovery is about to occur.

Table C-2 provides a list of confirmatory indicators for RCS inventory loss. Due to the variability of accuracy and usability of RVLIS while in Cold Shutdown or Refueling Mode, the use of RVLIS for emergency classification purposes is contingent on one or more of the listed confirmatory indications (ref. 3).

EAL RVLIS values have been rounded up to the nearest whole percentage point.

Reference(s):

1. OU-AA-200, "Shutdown Risk Management"
2. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"
3. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
4. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CG1.2 General Emergency

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

AND

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

- UNPLANNED increase in **any** Table C-1 sump or tank level of sufficient magnitude to indicate core uncovery
- Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery
- **Any** containment area radiation monitor reading > 3 R/hr (Refueling Mode)
- Erratic source range monitor indications

AND

Any Containment Challenge indication, Table C-3

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

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

Table C-1 Sumps/Tanks
• Reactor Containment Sump • Pressurizer Relief Tank (PRT) • Primary Drain Transfer Tank (PDTT) • Component Cooling (CC) Surge Tank • Refueling Water Storage Tank (RWST)

Table C-3 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • CTMT hydrogen concentration ≥4% • UNPLANNED increase in CTMT pressure

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions.

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:

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

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

The inability to monitor ~~(reactor vessel/RCS [*PWR*] or~~ RCS ~~[*BWR*])~~ level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels (Table C-1). Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the ~~(reactor vessel/~~RCS~~[*PWR*] or RPV [*BWR*])~~ (ref. 2, 3, 4).

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refueling mode, normal means of RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory.

In the Refueling mode, the dose rate above the core will rise as water level in the reactor vessel lowers. The dose rate due to this core shine should result in on-scale indications of > 3 R/hr on containment area radiation monitors (ref. 5).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.

~~In EAL 2.b, t~~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.

Three conditions are associated with a challenge to containment's capability to serve as an effective barrier to fission product release:

1. With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment (ref.1). If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

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

   In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. However, 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 burn (4%) (ref. 6). If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.

3. Any UNPLANNED rise in containment pressure in the Cold Shutdown or Refueling mode indicates a potential challenge of CONTAINMENT CLOSURE capability. This is due to the potential use of temporary penetration seals, water seals or other closure mechanisms used to support maintenance that are not suitable to withstand a rise in containment pressure. UNPLANNED containment pressure rise indicates

CONTAINMENT CLOSURE cannot be assured and the containment cannot be relied upon as a barrier to fission product release.

This~~ese~~ EAL~~s~~ 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. OU-AA-200, "Shutdown Risk Management"
2. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
3. 1(2)-OP-4.1, "Controlling Procedure for Refueling"
4. 1(2)-AP-17, "Shutdown LOCA"
5. RA-0078, "Verification of Radiation Monitor Response to Core Uncovery"
6. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"
7. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CU2.1 NOUE

AC power capability, Table C-4, to Unit 1(2) 4160V emergency buses H and J 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 SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.

Table C-4 AC Power Sources
Offsite: Unit 1 • Transfer Bus D • Transfer Bus F • Station Bus 1B • Station Bus 2B Unit 2 • Transfer Bus E • Transfer Bus F • Station Bus 2C • Station Bus 1A **Onsite:** • 1(2)H EDG • 1(2)J EDG • AAC (SBO) Diesel Generator (if already aligned)

Mode Applicability:

5 - Cold Shutdown, 6 - 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:

Table C-4 provides a list of offsite and onsite AC electrical power sources credited for this EAL.

Unit 1(2) 4160V emergency buses H and J are the emergency buses (ref. 5).

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

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

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

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

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

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

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down, the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs). The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). Additional bus ties for Unit 1exist between the 1H emergency bus to 1B station service bus and 1J emergency bus to 2B station service bus which can provide a second independent offsite power sources to each Unit 1 emergency bus. Unit 2 emergency busses can be cross tied between the following: 2C station service bus to 2H and 1A station service bus to 2J, which can provide a second independent offsite power source to each Unit 2 emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions. A bus that is powered from the SBO can be credited as being powered from an independent power source. However, since it takes longer than 15 minutes to align the SBO diesel generator, the generator must be "already aligned" to credit it as an AC power source.

Reference(s):

1. 11715-FE-1A, "Main One Line Diagram (Unit 1)"
2. 12050-FE-1A, "Main One Line Diagram (Unit 2)"
3. 1(2)-ECA-0.0, "Loss of All AC Power"
4. 0-AP-10, "Loss of Electrical Power"
5. UFSAR Section 8.3
6. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CA2.1	**Alert**
Loss of **all** offsite and **all** onsite AC power to Unit 1(2) 4160V emergency buses H and J for ≥15 min. (Note 1)	

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

Mode Applicability:

5 - Cold Shutdown, 6 - 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:

For this EAL credit can be taken for any AC power source that has sufficient capability to operate equipment necessary to maintain a safe shutdown condition, such as FLEX generators, provided it can be aligned within the 15 minute classification criteria.

Unit 1(2) 4160V emergency buses H and J are the emergency buses (ref. 5).

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

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

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

Escalation of the emergency classification level would be via ICs CS1 or ~~AS1~~RS1.

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down, the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs). The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). Additional bus ties for Unit 1exist between the 1H emergency bus to 1B station service bus and 1J emergency bus to 2B station service bus which can provide a second independent offsite power sources to each Unit 1 emergency bus. Unit 2 emergency busses can be cross tied between the following: 2C station service bus to 2H and 1A station service bus to 2J, which can provide a second independent offsite power source to each Unit 2 emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions. A bus that is powered from the SBO can be credited as being powered from an independent power source. However, since it takes longer than 15 minutes to align the SBO diesel generator, the generator must be “already aligned” to credit it as an AC power source.

This cold condition EAL is equivalent to the hot condition EAL MS1.1.

Reference(s):

1. 11715-FE-1A, “Main One Line Diagram (Unit 1)”
2. 12050-FE-1A, “Main One Line Diagram (Unit 2)”
3. 1(2)-ECA-0.0, “Loss of All AC Power”
4. 0-AP-10, “Loss of Electrical Power”
5. UFSAR Section 8.3
6. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

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

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions.

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 time to 200°F data when in Mode 6 or the RCS is not intact in Mode 5 (ref. 1). If the RCS is intact, classification should be based on the RCS pressure increase criteria of CA3.1. Guidance for calculating RCS time to 200°F is provided in 1(2)-AP-11 Loss of RHR (ref. 2).

This ~~IC~~ EAL addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit ~~or the inability to determine RCS temperature and level,~~and represents a potential degradation of the level of safety of the plant (ref. 1). If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the ~~Emergency Director~~SEM should also refer to ~~IC~~ 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.

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

During an outage, the level in the reactor vessel will normally be maintained at or above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid increase in reactor coolant temperature depending on the time after shutdown (ref. 2).

~~EAL #2 reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key~~

~~parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.~~

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

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

Reference(s):

1. Technical Specifications Table 1.1-1
2. 1(2)-AP-11, "Loss of RHR"
3. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

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

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

Mode Applicability:

5 - Cold Shutdown, 6- Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions.

Basis:

This ~~IC EAL~~EAL addresses ~~an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit, or~~ the inability to determine RCS temperature and level, ~~and~~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~~SEM should also refer to ~~IC~~ 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.~~

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

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

~~EAL #2This EAL~~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.

RCS level indications include (ref. 2):

- Standpipe level indication 1(2)-RC-LI-102

- Cold Shutdown Level Indicator 1(2)-RC-LI-103
- Independent RCS Level Indicator 1(2)-RC-LI-105
- RVLIS Upper Range Train
- RVLIS Full Range

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.

Reference(s):

1. Technical Specifications Table 1.1-1
2. 1(2)-OP-5.4, “Draining the Reactor Coolant System”
3. 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 increase in RCS temperature to > 200°F for > Table C-5 duration (Notes 1, 12)

OR

UNPLANNED RCS pressure increase > 10 psi (does not apply to solid plant conditions)

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

Note 12: If an RCS heat removal system is in operation within the applicable Table C-5 heat-up duration and RCS temperature is being reduced, the EAL is **not** applicable.

Table C-5 RCS Heat-up Duration Thresholds

RCS Status	CONTAINMENT CLOSURE Status	Heat-up Duration
Intact **AND not** reduced/decreased inventory		60 min.
Not intact **OR** reduced/decreased inventory	Established	20 min.
	Not established	0 min.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions.

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 time to 200°F data when in Mode 6 or the RCS is not intact in Mode 5 (ref. 1). If the RCS is intact, classification should be based on the RCS pressure increase criteria of CA3.1. Guidance for calculating RCS time to 200°F is provided in 1(2)-AP-11 Loss of RHR (ref. 2).

Decreased Inventory is defined as a condition with fuel in the Reactor Vessel and any RCS Loop Stop Valve closed, or RCS water level less than five percent (5%) in the pressurizer. With the Reactor Vessel Head removed and the Reactor Cavity filled to at least 23 feet above the Reactor Vessel Flange, the RCS is not considered to be in a decreased inventory condition (ref. 3).

Reduced Inventory is defined as a condition with fuel in the Reactor Vessel and water level lower than three feet below the Reactor Vessel flange. This corresponds to a plant elevation of 259.8 ft. If reading RCS Level from the MCR on 1(2)-RC-LI-102, RCS Standpipe, Reduced Inventory corresponds to an indicated level of 42 inches (ref. 3).

This ~~IC~~ 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 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, or RCS inventory is reduced (e.g., mid-loop operation in PWRs). 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 or is at reduced inventory ~~[*PWR*]~~, 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 should be assumed to be intact when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals). With the Pressurizer PORV(s) blocked open, the RCS is considered not intact.

The RCS pressure increase threshold ~~EAL #2~~ provides a pressure-based indication of RCS heat-up in the absence of RCS temperature monitoring capability. 1(2)-RC-PI-1403B and 1(2)-RC-PI-1402B provide RCS narrow range pressure indication (ref. 4, 5).

Escalation of the emergency classification level would be via IC CS1 or ~~AS1~~RS1.

Reference(s):

1. Technical Specifications Table 1.1-1
2. 1(2)-AP-11, "Loss of RHR"
3. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
4. 1-ICP-RC-P1403 (2-ICP-RC-P2403), "Reactor Coolant System Pressure (Wide and Narrow Range) Protection Channel IV Calibration"

5. 1-ICP-RC-P1402 (2-ICP-RC-P2402), "Reactor Coolant System Pressure (Wide and Narrow Range) Protection Channel I Calibration"
6. 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	**NOUE**
Indicated voltage is < 105 VDC on **required** vital 125 VDC battery buses for ≥15 min. (Note 1)	

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

Mode Applicability:

5 - Cold Shutdown, 6 - 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

There are four independent 125 volt DC systems for each unit.

Each system consists of 125 volt DC distribution panels and its respective battery and a battery charger. The batteries 1(2)-I, 1(2)-II, 1(2)-III, and 1(2)-IV supply power only if the battery chargers fail or if the demand exceeds the capacity of the chargers. The batteries are rated for a minimum of two hours (ref. 1, 2).

A battery terminal voltage of 105 volts DC is the minimum voltage required to ensure proper operation of equipment connected to the DC bus (ref. 4).

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

As used in this EAL, "required" means the vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if 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~~NOUE. A loss of vital DC power to Train A would not warrant an emergency classification.

The term "required" is meant to be consistent with the requirements of Technical Specifications for the plant shutdown operating modes (ref. 3).

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

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

This cold condition EAL is equivalent to the hot condition EAL MS2.1.

Reference(s):

1. 1(2)-AP-10, "Loss of Electrical Power"
2. UFSAR Section 8.3.2, "Direct Current Power System"
3. Technical Specifications Section 3.8.5, "DC Sources – Shutdown"
4. 0-OP-6.4, "Operation of the SBO Diesel (SBO Event)"
5. 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 NOUE

Loss of **all** Table C-6 onsite communication methods

OR

Loss of **all** Table C-6 State and local agency communication methods

OR

Loss of **all** Table C-6 NRC communication methods

Table C-6 Communication Methods

System	Onsite	State/ Local	NRC
Radio Communications System	X		
Public Address and Intercom System	X		
Private Branch Telephone Exchange (PBX)	X	X	X
Sound Powered Telephone System	X		
Commercial Telephone System		X	X
Automatic Ring Downs (SONET Ring)		X	
Instaphone Loop		X	
Dedicated NRC Communications			X

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, DEF – Defueled

Definition(s):

None

Basis:

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

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

The first EAL condition ~~EAL #1~~ addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition ~~EAL #2~~ addresses a total loss of the communications methods used to notify all ~~OROs~~ State and local agencies of an emergency declaration. The ~~OROs~~ State and local agencies referred to here are ~~(see Developer Notes)~~the Commonwealth of Virginia and affected local communities.

The third EAL condition ~~EAL #3~~ addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

This cold condition EAL is equivalent to the hot condition EAL MU7.1.

Reference(s):

1. North Anna Power Station Emergency Plan, Section 7.2, “Communications Systems”
2. UFSAR Section 9.5.2, “Communication Systems”
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 second train of the SAFETY SYSTEM needed for the current operating mode
- Event damage has resulted in VISIBLE DAMAGE to the second 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/SEM

Mode Applicability:

5 - Cold Shutdown, 6 - 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.

Basis:

This IC 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 second 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 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.

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

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

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

An event affecting equipment common to two or more trains of a safety system (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the common equipment) should be classified as an Alert under this EAL, as appropriate to the plant mode. By affecting the functionality of multiple trains of a safety system, the loss of the common equipment effectively meets the two-train impact criteria that underlie the EALs and bases.

An event affecting a single-train safety system (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under this EAL because the two-train impact criteria that underlie the EALs and bases would not be met. If an event affects a single-train safety system, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on SEM judgement.

An event that affects two trains of a safety system (e.g., one train has indications of degraded performance and the other VISIBLE DAMAGE) that also has one or more additional trains should be classified as an Alert under this EAL, as appropriate to the plant mode. This approach maintains consistency with the two-train impact criteria that underlie the EALs and bases and is warranted because the event was severe enough to affect the functionality of two trains of a safety system despite plant design criteria associated with system and system train separation and protection. Such an event may have caused other plant impacts that are not immediately apparent.

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

This cold condition EAL is equivalent to the hot condition EAL MA8.1.

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.

A NOUE 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 NAPS ISFSI is located outside the NAPS PLANT PROTECTED AREA but within the OWNER CONTROLLED AREA. Therefore a hostile security event that leads to a potential loss in the level of safety of the ISFSI is a classifiable event under Security category EAL HA4.1.

Category: ISFSI

Subcategory: Confinement Boundary

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 NOUE

Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask > **any** Table E-1 limit

Table E-1 ISFSI Cask Surface Dose Rate Limits

TN-32	TN-32B HBU	HSM-H
• 116 mrem/hr (neutron + gamma) average on top of the cask • 436 mrem/hr (neutron + gamma) average on the side of the cask	• 192 mrem/hr (neutron + gamma) average on top of the cask • 436 mrem/hr (neutron + gamma) average on the side of the cask	• 1,600 mrem/hr at the front bird screen • 4 mrem/hr at the door centerline • 4 mrem/hr at the end shield wall exterior

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY- The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the NAPS ISFSI, Confinement Boundary is defined as the Sealed Surface Storage Cask (SSSC) or 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 IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage.

The existence of "damage" is determined by radiological survey. The specified EAL threshold values correspond to 2 times the TN-32, TN-32B HBU or Horizontal Storage Module (HSM-H) external cask surface dose rate limits (ref. 1, 2). The technical specification multiple of "2 times", which is also used in ~~Recognition~~ Category ~~A~~ R IC ~~AU1~~RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

NAPS utilizes the Transnuclear TN-32/TN-32B HBU dry storage cask system and the NUHOMS HD System (32PTH DSC/HSM-H) dry cask storage system (ref 1).

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

Reference(s):

1. North Anna Power Station ISFSI NRC Certificate of Compliance 1030 Amendment 1, "Technical Specifications and SER (HSM-H)"
2. Technical Specifications and Bases for North Anna ISFSI (TN-32/TN-32B HBU)
3. NEI 99-01 E-HU1

Category F – Fission Product Barrier Degradation

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

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

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

B. Reactor Coolant System Barrier (RCS): The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

C. Containment Barrier (CTMT): The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the Emergency Classification Level (ECL) from an 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

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 Containment Barrier.
- ~~Unusual Event~~NOUE 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~~ RG1 has been exceeded.
- The fission product barrier thresholds specified within a scheme reflect plant-specific NAPS 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~~SEM would have more assurance that there was no immediate need to escalate to a General Emergency.

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **Any** loss or **any** potential loss of either Fuel Clad or RCS

EAL:

FA1.1	**Alert**
Any loss or **any** potential loss of **EITHER** Fuel Clad or RCS barrier (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 – Startup, 3 – Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

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

At the Alert classification level, Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1

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 Standby, 4 - 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 Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds, bases and references.

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

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

At the Site Area Emergency classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, they 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 Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

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

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

- Loss of Fuel Clad, RCS and Containment Barriers
- Loss of Fuel Clad and RCS Barriers with potential loss of Containment Barrier
- Loss of RCS and Containment Barriers with potential loss of Fuel Clad Barrier
- Loss of Fuel Clad and Containment Barriers with potential loss of RCS Barrier

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 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. RCS or SG Tube Leakage

B. Inadequate Heat removal

C. CTMT Radiation / RCS Activity

D. CTMT Integrity or Bypass

E. SEM 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.

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

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

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

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

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad Barrier (FC)		Reactor Coolant System Barrier (RCS)		Containment Barrier (CTMT)	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A RCS or SG Tube Leakage	None	None	1. An automatic or manual Safety Injection (SI) actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	1. UNISOLABLE RCS or SG tube leakage > 150 gpm 2. Integrity-RED Path conditions met	1. A leaking or RUPTURED SG is FAULTED outside of CTMT	None
B Inadequate Heat Removal	1. Core Cooling-RED Path conditions met	1. Core Cooling-**ORANGE** Path conditions met 2. Heat Sink-RED Path conditions met **AND** Heat sink is required	None	3. Heat Sink-RED Path conditions met **AND** Heat sink is required	None	1. Core Cooling-RED PATH conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C CTMT Radiation / RCS Activity	2. CTMT High Range Radiation Monitor RM-RMS-165/166(265/266) reading > Table F-2 column Fuel Clad Loss 3. Coolant activity > 300 μCi/gm DEI-131 4. Dose rate at 1 ft. from an unpressurized RCS sample ≥Table F-3 5. Sample line dose rate threshold ≥Table F-4 6. With letdown in service, Reactor Coolant Letdown Radiation Monitor CH-RI-128(228) > 7.5E+04 mR/hr	None	2. CTMT High Range Radiation Monitor RM-RMS-165/166(265/266) reading > Table F-2 column RCS Loss	None	None	2. CTMT High Range Radiation Monitor RM-RMS-165/166(265/266) reading > Table F-2 column CTMT Potential Loss
D CTMT Integrity or Bypass	None	None	None	None	2. CTMT isolation (Phase A or B) is required **AND EITHER:** • CTMT integrity has been lost based on SEM judgment • UNISOLABLE pathway from CTMT atmosphere to the environment exists 3. Indications of UNISOLABLE RCS leakage outside of CTMT	3. Containment-RED Path conditions met 4. CTMT hydrogen concentration ≥4% 5. CTMT pressure > 28 psia with < one full train of CTMT heat removal systems (Note 11) operating per design for ≥15 min. (Note 1)
E SEM Judgment	7. **Any** condition in the opinion of the SEM that indicates loss of the fuel clad barrier	3. **Any** condition in the opinion of the SEM that indicates potential loss of the fuel clad barrier	3. **Any** condition in the opinion of the SEM that indicates loss of the RCS barrier	4. **Any** condition in the opinion of the SEM that indicates potential loss of the RCS barrier	4. **Any** condition in the opinion of the SEM that indicates loss of the CTMT barrier	6. **Any** condition in the opinion of the SEM that indicates potential loss of the CTMT barrier

Barrier: Fuel Clad

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

1. Core Cooling-RED Path conditions met

Definition(s):

None

Basis:

This ~~reading~~ condition indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

The loss threshold is based on meeting either CSFST Core Cooling Red path criteria (ref. 1, 2):

- Core Exit Thermocouple readings ≥1,200 °F.
- Core exit TCs are ≥700°F with RCS subcooling based on core exit TCs ≤25°F [75°F], no RCPs are running, and RVLIS full range is ≤48%

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 2 Core Cooling"
2. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. Core Cooling-ORANGE Path conditions met

Definition(s):

None

Basis:

This ~~reading~~ condition indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

The potential loss threshold is based on meeting the CSFST Core Cooling Orange Path criteria.

CSFST Core Cooling-ORANGE path is entered if core exit thermocouples (TCs) are < 1,200°F, RCS subcooling based on core exit TCs is ≤25°F [75°F], and either of the following (ref. 1, 2):

- No RCPs are running and either: core exit TCs are ≥700°F and RVLIS full range is > 48%, or core exit TCs are < 700°F and RVLIS full range is ≤48%.
- At least one RCP is running and Reactor Vessel water level is ≤the specified RVLIS dynamic head threshold readings based on the number of RCPs running.

Reactor Vessel Water Level Thresholds		
RVLIS	**No. RCPs**	**Threshold**
Full Range	None	48%
Dynamic Range	3	65%
	2	41%
	1	30%

Consistent with Section 3.2.6 Classification of Transient Conditions, expected short term CSFST Core Cooling-ORANGE path conditions existing prior to successful automatic ECCS actuation following a large break LOCA would not meet the intent of this threshold.

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 2 Core Cooling"
2. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"

3. NEI 99-01 Inadequate Heat Removal Fuel Clad Potential Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. Heat Sink-RED Path conditions met **AND** Heat sink is required

Definition(s):

None

Basis:

The potential loss threshold is based on meeting the CSFST Heat Sink Red Path criteria of both of the following conditions existing (ref. 1):

- Narrow Range levels in all SGs < 11% [22%]
- Total feedwater flow to SGs $\leq$340 gpm

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

The phrase "and heat sink required" precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. For example, FR-H.1 is entered from CSFST Heat Sink-Red. Step 1 tells the operator to determine if secondary heat sink is required by checking that RCS pressure is greater than any non-faulted SG pressure and RCS T_{hot} is greater than 350°F. If these conditions exist, Heat Sink is required. Otherwise, the operator is to either go to the procedure and step in effect or place RHR in service for heat removal. For large LOCA events inside the Containment, the SGs are irrelevant because heat removal through the containment heat removal systems takes place. Therefore, Heat Sink Red should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification. (ref. 1, 2).

Meeting this threshold results in a Site Area Emergency because this threshold is identical to RCS Barrier Potential Loss threshold ~~2.A~~B.3; both will be met. This condition warrants a Site Area Emergency declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase RCS pressure to the point where mass will be lost from the system.

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 3 Heat Sink"
2. 1(2)-FR-H.1, "Response to Loss of Secondary Heat Sink"
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Potential Loss 2.B

Barrier: Fuel Clad

Category: C. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. CTMT high range radiation monitor RM-RMS-165/166(265/266) reading > Table F-2 column Fuel Clad Loss

Table F-2 CTMT High Range Radiation Monitor Barrier Thresholds RM-RMS-165/166(265/266)			
Time > Shutdown (hrs)	**Fuel Clad Loss (R/hr)**	**RCS Loss (R/hr)**	**CTMT Potential Loss (R/hr)**
≤2	125	5	500
> 2 – ≤4	85	5	340
> 4 – ≤6	45	5	180
> 8 – ≤14	20	5	80
> 14	10	5	40

Definition(s):

None

Basis:

Containment radiation monitor readings greater than the Table F-2 Fuel Clad Loss column threshold indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading is derived assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 5% clad failure into the containment atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within Technical Specifications and are therefore indicative of fuel damage (approximately 5 % clad failure depending on core inventory and RCS volume)~~The radiation monitor reading 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 an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier~~ (ref. 1, 2).

Time after shutdown values are provided to account for radioactive decay.

The values specified in Table F-2 were developed using a method to minimize error (+/-) for the threshold value within each defined time period. Time periods were chosen to fit monitor response (fast changes in response early following reactor shutdown are broken up into smaller time periods to better approximate expected change). Values were chosen within each time period to minimize error (<50%) to the highest and lowest response within the range.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold C.~~1~~2 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 RCS Activity / Containment Radiation.

Reference(s):

1. Calculation RA-0064, "Expected Containment High Range Radiation Monitor Response to a LOCA Based on Fuel Rod Gap Fractions Defined in NUREG 1228"
2. NEI 99-01 CTMT Radiation / RCS Activity FC Loss 3.A

Barrier: Fuel Clad

Category: C. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

3. Coolant activity > 300 µCi/gm DEI-131

Definition(s):

None

Basis:

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

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 RCS Activity / Containment Radiation.

Reference(s):

1. NEI 99-01 CTMT Radiation / RCS Activity FC Loss 3.B

Barrier: Fuel Clad

Category: C. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

4. Dose rate at 1 ft. from an unpressurized RCS sample ≥Table F-3

Table F-3 FC Loss Coolant Activity Dose Rates	
Time > Shutdown (hrs)	**mR/hr/ml**
≤2	15
> 2 – ≤8	8
> 8	3

Definition(s):

None

Basis:

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

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. <u>This EAL provides the ability to take a dose rate off of an RCS sample to determine fuel clad barrier loss, without the need to analyze the sample before making this determination. This EAL saves significant time by allowing evaluation of contained radioactivity within the RCS by a direct dose rate measurement.</u>

<u>Per Engineering Calculation RA-0059, dose rate is assumed to result from radioactive iodines (I-131 thru I-135) in RCS in concentrations corresponding to the loss of 5% of gap radioactivity of the core. For 5% loss of gap radioactivity (~300 μCi/gm DEI-131), 2% of the core inventory of radioactive iodines are assumed to be contained in the gap. The values contained in Table F-3 (FC Loss Coolant Activity Dose Rates) represent expected one foot dose rates per ml of sample based on time since reactor shutdown to the time when the sample is taken. The expected dose rate is a near linear relationship with the volume of the sample, so any volume collected can be determined by dividing the measured dose rate by the sample volume and comparing to the threshold value from Table F-3 for the applicable time frame. These dose rates assume no ECCS injection so there is no dilution credited which would vary coolant volume. Values in the table have been rounded for ease of use. The > 8 hour threshold is conservative up to 24 hours following reactor shutdown. After 24 hours, the expected</u>

response from radioactive iodine levels off. Therefore, the value shown for > 8 hours applies for all samples taken 8 hours or more since reactor shutdown (ref. 1, 2).

The values specified in Table F-3 were developed using a method to minimize error (+/-) for the threshold value within each defined time period. Values were chosen to minimize error from the highest to lowest dose rate within each range.

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

Reference(s):

1. Calculation RA-0059, "Detector Response to an RCS Sample for EAL Classification of Fuel Clad Degradation and Barrier Loss"
2. NEI 99-01 CTMT Radiation / RCS Activity FC Loss 3.B

Barrier: Fuel Clad

Category: C. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

5. Sample line dose rate threshold ≥Table F-4

Table F-4 FC Loss RCS Sample Line Dose Rates	
Time > Shutdown (hrs)	**R/hr**
≤2	4
> 2 – ≤8	2
> 8	1

Definition(s):

None

Basis:

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

Per Engineering Calculation RA-0079, dose rate is assumed to result from radioactive iodines in the RCS in concentrations corresponding to the loss of 5% of gap radioactivity of the core. The values contained in Table F-4 (FC Loss RCS Sample Line Dose Rates) represent fuel clad failure thresholds when measured approximately 2" from the outside of the RCS hot leg sample line. RCS sample line locations have been predetermined for use with this EAL. Other RCS lines could be used if analyzed on a case-by-case basis. Values in the table have been rounded for ease of use. The sample line dose rates have been calculated for various time ranges after shutdown (ref. 1).

~~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.~~

The values specified in Table F-4 were developed using a method to minimize error (+/-) for the threshold value within each defined time period. Values were chosen to minimize error from the highest to lowest dose rate within each range.

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

Reference(s):

1. Engineering Calculation RA-0079

2. NEI 99-01 CTMT Radiation / RCS Activity FC Loss 3.B

Barrier: Fuel Clad

Category: C. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

6. With letdown in service, Reactor Coolant Letdown Radiation Monitor 1(2)-CH-RI-128(228) > 7.5E+04 mrem/hr

Definition(s):

None

Basis:

This threshold indicates that RCS radioactivity concentration is greater than 300 μCi/gm DEI-131 (ref. 1). Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

~~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.~~

A portion of the letdown stream flows past radiation monitors 1(2)-CH-RM-128(228) to detect fission product activity in the reactor coolant and warn of a potential fuel element failure (ref. 2).

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

Reference(s):

1. Calculation No. PA-0234, Rev. 1 "Post Accident Letdown Radiation Monitor Response for North Anna"
2. UFSAR Section 11.4.2.15, "Reactor Coolant Letdown Gross Activity Monitors"
3. NEI 99-01 CTMT Radiation / RCS Activity FC Loss 3.B

Barrier: Fuel Clad

Category: C. CTMT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. SEM Judgment

Degradation Threat: Loss

Threshold:

7. **Any** condition in the opinion of the SEM 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~~SEM 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. SEM Judgment

Degradation Threat: Potential Loss

Threshold:

3. **Any** condition in the opinion of the SEM 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~~SEM in determining whether the Fuel Clad barrier is potentially lost. The ~~Emergency Director~~SEM 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. RCS or S/G Tube Leakage

Degradation Threat: Loss

Threshold:

1. An automatic or manual Safety Injection (SI) actuation required by **EITHER**:
   - UNISOLABLE RCS leakage
   - SG tube RUPTURE

Definition(s):

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

RUPTURE - The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

Basis:

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

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

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

This threshold does not apply to a Safety Injection (SI) actuation not caused by excessive RCS leakage (i.e., steamline ΔP or high steam flow) (ref. 1).

If EOPs direct operators to open the Pressurizer pressure relief valves to implement a core cooling strategy (i.e., a “feed and bleed” cooldown), then there will exist a reactor coolant flow path from the RCS, past the “pressurizer safety and relief valves” and into the containment that operators cannot isolate without compromising the effectiveness of the strategy (i.e., for the strategy to be effective, the valves must be kept in the open position); therefore, the flow through the pressure relief line is UNISOLABLE. In this case, the ability of the RCS pressure boundary to serve as an effective barrier to a release of fission products has been eliminated and thus this condition constitutes a loss of the RCS barrier.

Reference(s):

1. 1(2)-E-0, “Reactor Trip or Safety Injection”
2. 1(2)-E-3, “Steam Generator Tube Rupture”

3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Loss 1.A

Barrier: Reactor Coolant System

Category: A. RCS or S/G Tube Leakage

Degradation Threat: Potential Loss

Threshold:

1. UNISOLABLE RCS or SG tube leakage > 150 gpm

Definition(s):

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

Basis:

This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging ~~(makeup)~~ pump, but an ~~ECCS (SI)~~ actuation has not occurred. The threshold is met when RCS leakage is determined to exceed 150 gpm excluding normal reductions in RCS inventory such as letdown and RCP seal leakoff~~an operating procedure, or operating crew supervision, directs that a standby charging (makeup) pump be placed in service to restore and maintain pressurizer level~~ (ref.1)~~The threshold is met when an operating procedure, or operating crew supervision, directs that a standby charging (makeup) pump be placed in service to restore and maintain pressurizer level~~.

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

If ~~a~~ the leaking steam generator (> 150 gpm) is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold ~~1.A~~A.1 will also be met.

Reference(s):

1. NAPS FSAR Table 9.3-5, "Principal Component Data Summary"
2. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.A

Barrier: Reactor Coolant System

Category: A. RCS or S/G Tube Leakage

Degradation Threat: Potential Loss

Threshold:

2. Integrity-RED Path conditions met

Definition(s):

None

Basis:

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

The potential loss threshold is defined by the CSFST Integrity - RED path. CSFST Integrity - Red Path plant conditions (> 100°F/hr cold leg cooldown) and associated PTS Limit A Curve indicates an extreme challenge to the safety function when plant parameters are to the left of the limit curve following excessive RCS cooldown under pressure (ref. 1).

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees Attachment 4 Integrity"
2. 1(2)-FR-P.1, "Response to Imminent Pressurized Thermal Shock Condition"
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.B

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. Heat Sink-RED Path conditions met

   AND

   Heat sink is required

Definition(s):

None

Basis:

The potential loss threshold is based on meeting the CSFST Heat Sink Red Path criteria of both of the following conditions existing (ref. 1):

- Narrow Range levels in all SGs < 11% [22%]
- Total feedwater flow to SGs ≤340 gpm

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

The phrase "and heat sink required" precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. For example, FR-H.1 is entered from CSFST Heat Sink-Red. Step 1 tells the operator to determine if heat sink is required by checking that RCS pressure is greater than any non-faulted SG pressure and RCS T_{hot} is greater than 350°F. If these conditions exist, Heat Sink is required. Otherwise, the operator is to either go to the procedure and step in effect or place RHR in service for heat removal. For large LOCA events inside the Containment, the SGs are irrelevant because heat removal through the containment heat removal systems takes place. Therefore, Heat Sink Red is not applicable and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification. (ref. 1, 2).

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

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees Attachment 3 Heat Sink"
2. 1(2)-FR-H.1, "Response to Loss of Secondary Heat Sink"
3. NEI 99-01 Inadequate Heat Removal RCS Loss 2.B

Barrier: Reactor Coolant System

Category: C. CTMT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

2. CTMT high range radiation monitor RM-RMS-165/166(265/266) reading > Table F-2 column RCS Loss

Table F-2 CTMT High Range Radiation Monitor Barrier Thresholds RM-RMS-165/166(265/266)

Time > Shutdown (hrs)	Fuel Clad Loss (R/hr)	RCS Loss (R/hr)	CTMT Potential Loss (R/hr)
≤2	125	5	500
> 2 – ≤4	85	5	340
> 4 – ≤6	45	5	180
> 8 – ≤14	20	5	80
> 14	10	5	40

Definition(s):

None

Basis:

A reading > 5 R/hr (minimum practical reading) on RM-RMS-165/166(265/266) is indicative of a breach in the RCS barrier (ref. 1, 2).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad barrier loss threshold ~~3.A~~C.2 since it indicates a loss of the RCS Barrier only.

Because of the very high fuel clad integrity, only small amounts of noble gases would be dissolved in the primary coolant. Conservative estimates indicated that the readings from release of the normal RCS inventory would be below normal readings on the monitor while the station was operating. Therefore, a value 5 times the normal containment radiation monitor RM-RMS-165/166(265/266) reading of ~ 1 R/hr is used. The reading is less than that specified for fuel cladding barrier loss because no damage to the fuel cladding is assumed. Only leakage from the RCS is assumed for this barrier loss threshold. The value is high enough to preclude erroneous classification of barrier loss due to normal plant operations and is the lowest readable value on the monitors (ref. 1).

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

Reference(s):

1. Calculation RA-0064, "Expected Containment High Range Radiation Monitor Response to a LOCA Based on Fuel Rod Gap Fractions Defined in NUREG 1228"
2. NEI 99-01 CMT Radiation / RCS Activity RCS Loss 3.A

Barrier: Reactor Coolant System

Category: C. CTMT Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. SEM Judgment

Degradation Threat: Loss

Threshold:

3. **Any** condition in the opinion of the SEM 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~~SEM 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: E. SEM Judgment

Degradation Threat: Potential Loss

Threshold:

4. **Any** condition in the opinion of the SEM 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~~SEM in determining whether the RCS barrier is potentially lost. The ~~Emergency Director~~SEM 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: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. A leaking or RUPTURED SG is FAULTED outside of CTMT

Definition(s):

FAULTED - The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.

RUPTURED - The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

Basis:

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

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

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC ~~SU4~~ MU4 for the fuel clad barrier (i.e., RCS activity values) and IC ~~SU5~~ MU5 for the RCS barrier (i.e., RCS leak rate values).

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

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

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

The emergency classification levels resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

	Affected SG is FAULTED Outside of Containment?	
P-to-S Leak Rate	**Yes**	**No**
Less than or equal to 25 gpm	No classification	No classification
Greater than 25 gpm	~~Unusual Event~~NOUE per ~~SU4~~MU5.1	~~Unusual Event~~NOUE per ~~SU4~~MU5.1
~~Requires operation of a standby charging (makeup) pump~~> 150 gpm (*RCS Barrier Potential Loss*)	Site Area Emergency per FS1.1	Alert per FA1.1
Requires an automatic or manual ECCS (SI~~AS~~) actuation (*RCS Barrier Loss*)	Site Area Emergency per FS1.1	Alert per FA1.1

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

Reference(s):

1. 1-E-2 (2-E-2), "Faulted Steam Generator Isolation"
2. 1-E-3 (2-E-3), "Steam Generator Tube Rupture"
3. NEI 99-01 RCS or SG Tube Leakage Containment Loss 1.A

Barrier: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

None

Barrier: Containment

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. Core Cooling-RED Path conditions met

   AND

   Restoration procedures **not** effective within 15 min. (Note 1)

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

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:

The potential loss threshold is based on meeting either CSFST Core Cooling Red Path criteria (ref. 1, 2):

- Core Exit Thermocouple readings ≥1,200 °F.
- Core exit TCs are ≥700°F with RCS subcooling based on core exit TCs ≤25°F [75°F], no RCPs are running, and RVLIS full range is ≤48%

and restoration procedures not effective within 15 minutes.

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

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

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

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 2 Core Cooling"
2. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"
3. NEI 99-01 Inadequate Heat Removal Containment Potential Loss 2.A

Barrier: Containment

Category: C. CTMT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: C. CTMT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

2. CTMT high range radiation monitor RM-RMS-165/166(265/266) reading > Table F-2 column CTMT Potential Loss

Table F-2 CTMT High Range Radiation Monitor Barrier Thresholds RM-RMS-165/166(265/266)			
Time > Shutdown (hrs)	**Fuel Clad Loss (R/hr)**	**RCS Loss (R/hr)**	**CTMT Potential Loss (R/hr)**
≤2	125	5	500
> 2 – ≤4	85	5	340
> 4 – ≤6	45	5	180
> 8 – ≤14	20	5	80
> 14	10	5	40

Definition(s):

None

Basis:

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds (ref. 1).

Time after shutdown values are provided to account for radioactive decay.

The values specified in Table F-2 were developed using a method to minimize error (+/-) for the threshold value within each defined time period. Time periods were chosen to fit monitor response (fast changes in response early following reactor shutdown are broken up into smaller time periods to better approximate expected change). Values were chosen within each time period to minimize error (<50%) to the highest and lowest response within the range.

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.

Reference(s):

1. Calculation RA-0064, “Expected Containment High Range Radiation Monitor Response to a LOCA Based on Fuel Rod Gap Fractions Defined in NUREG 1228”
2. NEI 99-01 CMT Radiation / RCS Activity Containment Potential Loss 3.A

Barrier: Containment

Category: D. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

2. CTMT isolation (Phase A or B) is required

   AND EITHER:

   - CTMT integrity has been lost based on SEM judgment
   - UNISOLABLE pathway from CTMT atmosphere to the environment exists

Definition(s):

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

Basis:

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold ~~1.A~~A.1. Therefore this threshold is not applicable to steam generator tube leakage.

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

~~4.A.1~~First Threshold – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the ~~Emergency Director~~SEM will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure ~~9-F-4~~1. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

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

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

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

Refer to the top piping run of Figure ~~9-F-4~~1. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

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

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

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

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

Reference(s):

1. UFSAR Section 6.2.4, "Containment Isolation System"
2. NEI 99-01 CMT Integrity or Bypass Containment Loss 4.A

Barrier: Containment

Category: D. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

3. Indications of UNISOLABLE RCS leakage outside of CTMT

Definition(s):

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

Basis:

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

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Containment Loss Threshold A.1. Therefore this threshold is not applicable to steam generator tube leakage.

This threshold **does not** apply to an UNISOLABLE RSHX tube leak outside containment. Such leaks are properly addressed under the category R radiological release based EALs.

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

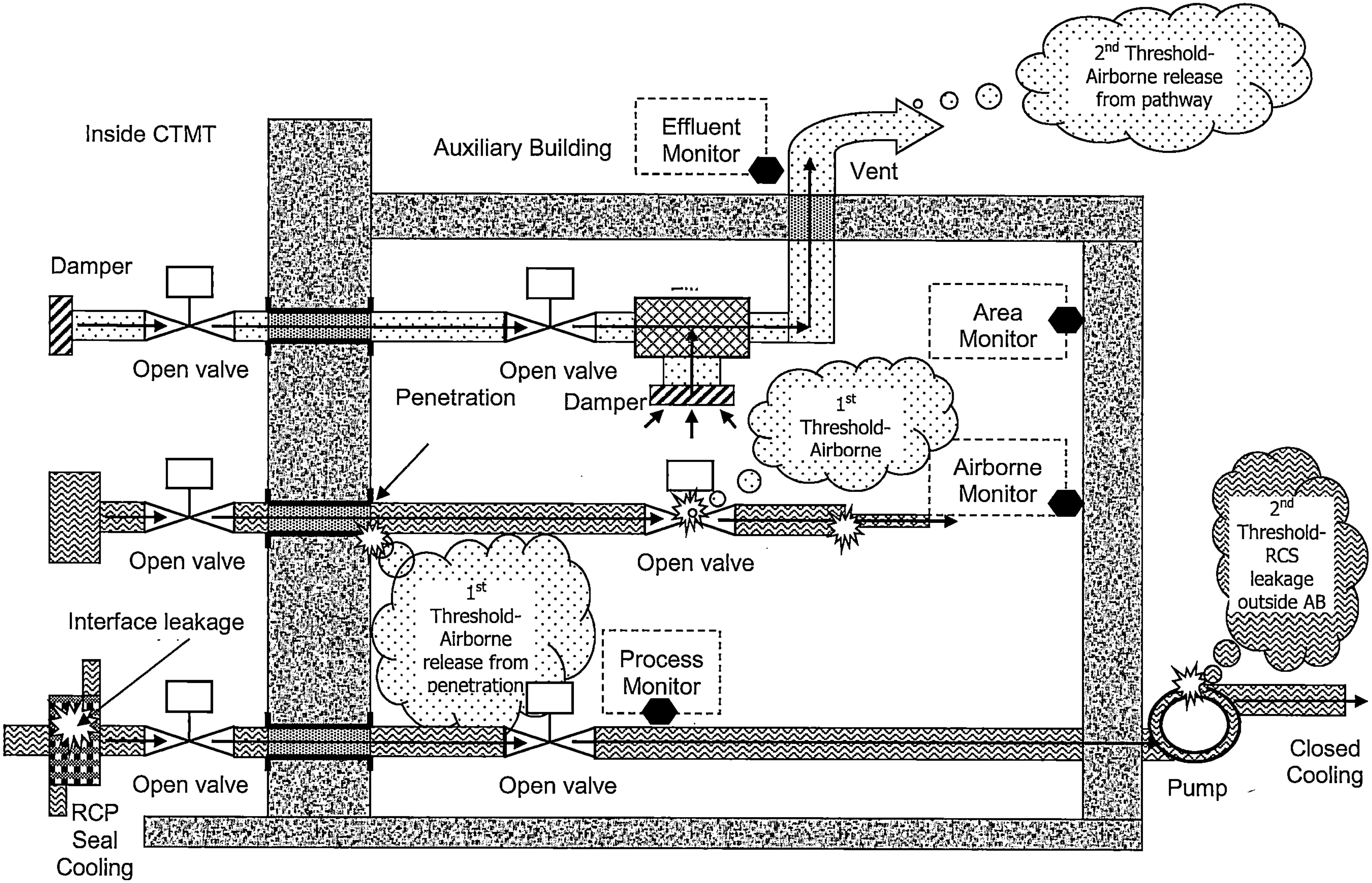
Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of RCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside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

Refer to the middle piping run of Figure ~~9-F-4~~1. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause loss threshold ~~4.A.1~~D.2 to be met as well.

Reference(s):

1. NEI 99-01 CMT Integrity or Bypass Containment Loss 4.B

Figure 1: Containment Integrity or Bypass Examples



Barrier: Containment

Category: D. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

3. Containment RED Path conditions met.

Definition(s):

None

Basis:

CSFST Containment RED Path conditions are met if containment pressure exceeds its design pressure. If containment pressure exceeds the design pressure of 60 psia (ref. 1, 2), there exists a potential to lose the containment barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the RCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a Site Area Emergency and General Emergency since there is now a potential to lose the third barrier.

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 5 Containment"
2. UFSAR Section 6.2
3. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.A

Barrier: Containment

Category: D. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

4. CTMT hydrogen concentration ≥4%

Definition(s):

None

Basis:

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

<u>A containment hydrogen concentration of 4% conservatively represents the lowest threshold for flammability in the presence of oxygen (ref. 1,2).</u>

<u>Containment hydrogen analyzers 1-HC-H2A-101 and 2-HC-H2A-201 display hydrogen concentration on PAMC-1 and PAMC-2 with a range of 0 - 10% (ref. 3).</u>

Reference(s):

1. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"
2. SAMG CA-3, "Calculation Aid Number 3 - Hydrogen Flammability in Containment:\"
3. UFSAR Table 7.5-2
4. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.B

Barrier: Containment

Category: D. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

5. CTMT pressure > 28 psia with < one full train of CTMT depressurization equipment (Note 11) operating per design for ≥15 min. (Note 1)

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

Note 11: One full train of containment depressurization equipment consist of one Quench Spray (QS) System and one Recirculation Spray (RS) System from either train operating together.

Definition(s):

None

Basis:

This threshold describes a condition where containment pressure is greater than the setpoint (28 psia) (ref. 3, 4) at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design (ref. 1, 2). The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems (e.g., containment sprays~~, ice condenser fans, etc.,~~ but not including containment venting strategies) are either lost or performing in a degraded manner.

The Quench Spray (QS) System, operating in conjunction with the Recirculation Spray (RS) System, is designed to cool and depressurize the containment structure to less than 2.0 psig in one hour and sub-atmospheric pressure in less than 6 hours following a Design Basis Accident. The combination of required equipment can be obtained from using equipment on either emergency busses in order to meet the "one full train" requirement (ref. 1, 2).

Reference(s):

1. Technical Specifications Section B 3.6.6, "Quench Spray (QS) System"
2. Technical Specifications Section B 3.6.,7 "Recirculation Spray (RS) System"
3. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 5 Containment"
4. 1(2)-FR-Z.1, "Response to High Containment Pressure"
5. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.C

Barrier: Containment

Category: E. SEM Judgment

Degradation Threat: Loss

Threshold:

4. **Any** condition in the opinion of the SEM that indicates loss of the CTMT barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that may be used by the ~~Emergency Director~~SEM in determining whether the containment barrier is lost.

Reference(s):

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

Barrier: Containment

Category: E. SEM Judgment

Degradation Threat: Potential Loss

Threshold:

6. **Any** condition in the opinion of the SEM that indicates potential loss of the CTMT barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that may be used by the ~~Emergency Director~~SEM in determining whether the containment barrier is potentially lost. The ~~Emergency Director~~SEM 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 Containment 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 PLANT 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.

7. SEM 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 SEM the latitude to classify emergency conditions consistent with the established classification criteria based upon SEM judgment.

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1 NOUE

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

<u>OR</u>

Notification of a credible security threat directed at the site

<u>OR</u>

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 NAPS 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 NAPS. 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) - The entire area contiguous to the PLANT PROTECTED AREA, owned by the Company and designated to be controlled for security reasons.

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

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

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.

Basis:

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR 73.71 or 10 CFR 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1~~,~~ and HS1 ~~and HG1~~. Guidance on assessing Security Conditions is included in the Security Contingency Implementing Procedures (SCIP). The SCIPs are implementing procedures for the Station Safeguards Contingency Plan.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1, 2, 3). Classification of these events will initiate appropriate threat-related notifications to plant personnel and ~~OROs~~State and local agencies.

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

The second threshold ~~EAL #2~~ addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the Millstone, North Anna and Surry Power Stations' Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program~~(site-specific procedure)~~ (ref. 1) and associated Security Plan Implementing Procedures (SCIP).

The third threshold ~~EAL #3~~ addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with 0-AP-9 Station Security 9 – Operations Response or 0-AP-9.01 Station Security Air Threat – Operations Response (ref. 2, 3)~~(site-specific procedure)~~.

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

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

Reference(s):

1. Millstone, North Anna and Surry Power Stations' Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program
2. 0-AP-9, "Station Security – Operations Response"
3. 0-AP-9.01, "Station Security Air Threat – Operations Response"
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HA1.1 Alert
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by NAPS 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 NAPS 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 NAPS. 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 - The entire area contiguous to the PLANT PROTECTED AREA, owned by the Company and designated to be controlled for security reasons.

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

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PLANT 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 (ref. 1, 2, 3).

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 State and local agencies~~Offsite Response Organizations~~, allowing them to be better prepared should it be necessary to consider further actions.

This ~~IC~~ 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 10 CFR 73.71 or 10 CFR 50.72.

The first threshold ~~EAL #1~~ is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes any action directed against an ISFSI that is located outside the PLANT PROTECTED AREA such as NAPS.

The second threshold ~~EAL #2~~ addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and State and local agencies~~OROs~~ are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with 0-AP-9 Station Security – Operations Response or 0-AP-9.01 Station Security Air Threat – Operations Response (ref. 2, 3) ~~(site-specific procedure)~~.

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

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

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

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

Reference(s):

1. Millstone, North Anna and Surry Power Stations' Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program
2. 0-AP-9, "Station Security – Operations Response"
3. 0-AP-9.01, "Station Security Air Threat – Operations Response"
4. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the PLANT PROTECTED AREA

EAL:

HS1.1	**Site Area Emergency**
A HOSTILE ACTION is occurring or has occurred within the PLANT PROTECTED AREA as reported by NAPS 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 NAPS 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 NAPS. 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 - The entire area contiguous to the PLANT PROTECTED AREA, owned by the Company and designated to be controlled for security reasons.

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

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

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the PLANT 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, 2, 3).

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 State and local agency~~ORO~~ resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

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

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

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

Reference(s):

1. Millstone, North Anna and Surry Power Stations' Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program
2. 0-AP-9, "Station Security – Operations Response"
3. 0-AP-9.01, "Station Security Air Threat – Operations Response"
4. 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	**NOUE**
Seismic event > OBE (0.06g horizontal or 0.04g vertical) as indicated by "OBE EXCEEDED" indicator illuminated on the SYSCOM Network Control Center (NCC)	

Mode Applicability:

All

Definition(s):

None

Basis:

0-AP-36 Seismic Event provides the guidance for determining if the OBE earthquake threshold is exceeded (horizontal or vertical) and any required response actions. (ref. 2).

Ground motion acceleration of 0.06g horizontal or 0.04g vertical is the Operating Basis Earthquake for NAPS (ref. 1).

Ground motion acceleration at the OBE is unmistakably a "felt" earthquake and is significantly greater than the ground motion acceleration required to activate the Event Indicator on the Strong Motion Accelerograph (SMA) which, in turn, activates annunciator 1A-B4, Earthquake System Trigger, in the Control Room. The "OBE EXCEEDED" indicator illuminates on the SYSCOM Network Control Center (NCC) if site OBE ground acceleration is exceeded (ref. 2).

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 significant seismic event (e.g., lateral accelerations in excess of 0.~~08g~~06g). 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.

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE). An earthquake greater than an OBE but less than a ~~Safe Shutdown~~Design Basis Earthquake (~~SSE~~DBE) 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.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or ~~SA9~~MA9.

Reference(s):

1. UFSAR Section 2.5.2.6
2. 0-AP-36, "Seismic Event"
3. 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	**NOUE**
A tornado strike within the PLANT PROTECTED AREA	

Mode Applicability:

All

Definition(s):

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

Basis:

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

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

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

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

~~EAL #4 addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site flooding caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.~~

~~This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.~~

~~EAL #5 addresses (site-specific description).~~

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

If damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under IC CA6 or MA9.

A tornado striking (touching down) within the PLANT PROTECTED AREA warrants declaration of an NOUE 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. 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	**NOUE**
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.

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

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 (ref. 1, 2).

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

~~EAL #4 addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site flooding caused by a hurricane, heavy rains,~~

~~up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.~~

~~This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.~~

~~EAL #5 addresses (site-specific description).~~

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

Refer to EAL CA6.1 or MA9.1 for internal flooding affecting more than one SAFETY SYSTEM train.

Reference(s):

1. 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	**NOUE**
Movement of personnel within the PLANT PROTECTED AREA is IMPEDED due to an event external to the PLANT 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).

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

Basis:

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

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

~~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, operabilit of the affected component must be required by Technical Specifications for the current operating mode.~~

~~EAL #3~~ addresses a hazardous materials event originating at a~~n~~ ~~offsite~~ location outside the PLANT PROTECTED AREA and of sufficient magnitude to IMPEDE the movement of personnel within the PLANT PROTECTED AREA.

~~EAL #4 addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site flooding caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.~~

~~This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane~~

~~Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.~~

~~EAL #5 addresses (site-specific description).~~

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

Reference(s):

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

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

Mode Applicability:

All

Definition(s):

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

~~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.~~

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

This EAL ~~EAL #4~~ addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site FLOODING caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.

This EAL is not intended 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.

~~EAL #5 addresses (site-specific description).~~ Escalation of the emergency classification level would be based on ICs in ~~Recognition~~ Categories ~~A~~R, F, ~~S~~ M or C.

Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.1 NOUE

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

Table H-1 NAPS Fire Areas
• Cable Vaults & Tunnels
• Emergency Switchgear Rooms
• Emergency Diesel Generator Rooms
• Reactor Containment
• Quench Spray Pump Houses
• Safeguards Area
• Main Steam Valve House
• Cable Spreading Rooms
• Control Room
• CR Chiller Rooms
• Auxiliary / Fuel / Decontamination Buildings
• Fuel Oil Pump House Room A or B
• Service Water Pump House and Valve House
• Intake Structure Control House
• Auxiliary Service Water Pump House
• Auxiliary Feedwater Pump House
• Turbine Building

Mode Applicability:

All

Definition(s):

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

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.

~~EAL #1~~

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.

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

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

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report. ~~EAL #2~~

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

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

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

~~EAL #3~~

~~In addition to a FIRE addressed by EAL #1 or EAL #2, a FIRE within the plant PROTECTED AREA not extinguished within 60 minutes may also potentially degrade the level of plant safety. This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA. [Sentence for plants with an ISFSI outside the plant Protected Area]~~

~~EAL #4~~

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

~~Basis-Related Requirements from Appendix R~~

~~Appendix R to 10 CFR 50, states in part:~~

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

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

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or ~~SA9~~MA9.

Reference(s):

1. NAPS Appendix R Report, Section 4.4 Attachment to Table 4-1
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 NOUE

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 (excluding Reactor Containment)

AND

The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Notes 1, 13)

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

Note 13: A Reactor Containment fire alarm is considered VALID upon receipt of multiple (more than one) fire zone alarms.

Table H-1 NAPS Fire Areas
• Cable Vaults & Tunnels
• Emergency Switchgear Rooms
• Emergency Diesel Generator Rooms
• Reactor Containment
• Quench Spray Pump Houses
• Safeguards Area
• Main Steam Valve House
• Cable Spreading Rooms
• Control Room
• CR Chiller Rooms
• Auxiliary / Fuel / Decontamination Buildings
• Fuel Oil Pump House Room A or B
• Service Water Pump House and Valve House
• Intake Structure Control House
• Auxiliary Service Water Pump House
• Auxiliary Feedwater Pump House
• Turbine Building

Mode Applicability:

All

Definition(s):

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

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.

~~EAL #1~~

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

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

~~EAL #2~~

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).

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

With regard to Reactor Containment fire alarms, there is constant air movement in the enclosed containment due to the operation of the containment ventilation system. The operating cooling units are drawing air to the units past the smoke detectors. It can be reasonably expected that a fire that burns for 15 minutes would produce sufficient products of combustion to cause fire detectors in multiple zones to alarm. Therefore, a single Reactor Containment fire alarm is not considered VALID.

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 ~~EAL #1~~ is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted. ~~EAL #3~~

~~In addition to a FIRE addressed by EAL #1 or EAL #2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety.~~ ~~*This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA.* [*Sentence for plants with an ISFSI outside the plant Protected Area*]~~

~~EAL #4~~

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

Basis-Related Requirements from Appendix R (justification for the use of 30 minute criteria)

~~Appendix R to~~ 10 CFR 50, Appendix R states in part:

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

In addition, ~~Appendix R to~~ 10 CFR 50, Appendix R, 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~~EAL #2~~, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or ~~SA9~~MA9.

Reference(s):

1. NAPS Appendix R Report, Section 4.4 Attachment to Table 4-1
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	**NOUE**
A FIRE within the PLANT PROTECTED AREA or ISFSI Protected Area **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)	

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

Mode Applicability:

All

Definition(s):

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

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

Basis:

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

~~EAL #1~~

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

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

~~EAL #2~~

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

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

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

~~EAL #3~~

In addition to a FIRE addressed by EAL HU4.1 ~~#1~~ or HU4.2~~EAL #2~~, a FIRE within the PLANT PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety.

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

~~Basis-Related Requirements from Appendix R~~

~~Appendix R to 10 CFR 50, states in part:~~

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

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

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or ~~SA9~~MA9.

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	**NOUE**
A FIRE within the PLANT PROTECTED AREA or ISFSI Protected Area that requires an offsite fire department to assist with extinguishment	

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.

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

Basis:

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

~~EAL #1~~

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

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

~~EAL #2~~

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

~~A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to~~

~~determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.~~

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

~~EAL #3~~

~~In addition to a FIRE addressed by EAL #1 or EAL #2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety.~~ ~~*This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA.*~~ ~~[*Sentence for plants with an ISFSI outside the plant Protected Area*]~~

~~EAL #4~~

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

The Shift Fire Brigade Incident Commander will assess whether the fire conditions warrant outside assistance (ref. 1).

~~Basis-Related Requirements from Appendix R~~

~~Appendix R to 10 CFR 50, states in part:~~

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

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

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or ~~SA9~~MA9.

Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gases

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

EAL:

HA5.1 Alert

Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 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**
Aux. Building El 274'	1, 2, 3, 4
Instrument Rack Rooms	4
Cable Vault & Tunnels	

Mode Applicability:

1 - Power Operation, 2 – Startup, 3 – Hot Standby, 4 - Hot Shutdown

Definition(s):

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

Basis:

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

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

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

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

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or IMPEDE a required action.
- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the 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~~. ~~(BWR only)~~.

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

Reference(s):

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

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

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

EAL:

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

Mode Applicability:

All

Definition(s):

None

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

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

Control will be established at the Auxiliary Shutdown Panel if the Control Room is evacuated for any reason (ref. 1, 2).

Escalation of the emergency classification level would be via IC HS6.

Reference(s):

1. 1(2)-AP-20, "Operation from the Auxiliary Shutdown Panel"
2. 0-FCA-1, "Control Room Fire"
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 has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel

AND

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

- Reactivity (modes 1, 2 and 3 **only**)
- Core cooling
- RCS heat removal

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

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling

Definition(s):

None

Basis:

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

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on ~~Emergency Director~~SEM judgment. The ~~Emergency Director~~SEM is expected to make a reasonable, informed judgment within 15 ~~(the site-specific time for transfer)~~ 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.

Control will be established at the Auxiliary Shutdown Panel if the Control Room was evacuated for any reason (ref. 1, 2).

Establishment of the reactivity safety function is only applicable in Modes 1, 2 and 3. Sufficient shutdown margin has already been established once in modes 4, 5 and 6 (ref.3).

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

Reference(s):

1. 1(2)-AP-20, “Operation from the Auxiliary Shutdown Panel”
2. 0-FCA-1, “Control Room Fire”
3. NRC EP FAQ 2015-014
4. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEM Judgment

Initiating Condition: Other conditions existing that in the judgment of the SEM warrant declaration of a NOUE

EAL:

HU7.1 NOUE

Other conditions exist which in the judgment of the SEM 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~~SEM to fall under the emergency classification level description for a NOUE.

Reference(s):

1. NEI 99-01 HU7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEM Judgment

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

EAL:

HA7.1	**Alert**

Other conditions exist which, in the judgment of the SEM, 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 NAPS 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 NAPS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

OWNER CONTROLLED AREA - The entire area contiguous to the PLANT PROTECTED AREA, owned by the Company and designated to be controlled for security reasons.

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

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

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~~SEM 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 – SEM Judgment

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

EAL:

HS7.1	**Site Area Emergency**

Other conditions exist which in the judgment of the SEM 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 NAPS 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 NAPS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

OWNER CONTROLLED AREA - The entire area contiguous to the PLANT PROTECTED AREA, owned by the Company and designated to be controlled for security reasons.

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

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

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~~SEM 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 – SEM Judgment

Initiating Condition: Other conditions exist that in the judgment of the SEM warrant declaration of a General Emergency

EAL:

HG7.1	**General Emergency**
Other conditions exist which in the judgment of the SEM 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.

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.

HOSTILE ACTION - An act toward NAPS 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 NAPS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

OWNER CONTROLLED AREA - The entire area contiguous to the PLANT PROTECTED AREA, owned by the Company and designated to be controlled for security reasons.

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

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

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~~SEM to fall under the emergency classification level description for a GENERAL EMERGENCY.

Reference(s):

1. NEI 99-01 HG7

Category M – System Malfunction

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

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

The events of this category pertain to the following subcategories:

1. Loss of Emergency AC Power

Loss of emergency 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 4160V emergency 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 increase from these base-line levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

5. RCS Leakage

The reactor 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 trips. In the plant licensing basis, postulated failures of the RPS to complete a reactor trip 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 properly result in 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. Containment Failure

Failure of containment isolation capability (under conditions in which the containment is not currently challenged) warrants emergency classification. Failure of containment pressure control capability also warrants emergency classification.

9. Hazardous Event Affecting Safety Systems

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

Category: M – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

MU1.1 NOUE

Loss of **all** offsite AC power capability, Table M-1, to Unit 1(2) 4160V emergency buses H and J for ≥15 min. (Note 1)

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

Table M-1 AC Power Sources
Offsite: Unit 1 • Transfer Bus D • Transfer Bus F • Station Bus 1B • Station Bus 2B Unit 2 • Transfer Bus E • Transfer Bus F • Station Bus 2C • Station Bus 1A
Onsite: • 1(2)H EDG • 1(2)J EDG • AAC (SBO) Diesel Generator (if already aligned)

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

Basis:

Table M-1 provides a list of offsite AC electrical power sources credited for this EAL.

Unit 1(2) 4160V emergency buses H and J are the emergency buses (ref. 1).

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

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

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

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down, the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs). The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). Additional bus ties for Unit 1 exist between the 1H emergency bus to 1B station service bus and 1J emergency bus to 2B station service bus which can provide a second independent offsite power sources to each Unit 1 emergency bus. Unit 2 emergency busses can be cross tied between the following: 2C station service bus to 2H and 1A station service bus to 2J, which can provide a second independent offsite power source to each Unit 2 emergency bus.

Escalation of the emergency classification level would be via IC ~~SA1~~MA1.

Reference(s):

1. 11715-FE-1A, "Main One Line Diagram (Unit 1)"
2. 12050-FE-1A, "Main One Line Diagram (Unit 2)"
3. 1(2)-ECA-0.0, "Loss of All AC Power"
4. 0-AP-10, "Loss of Electrical Power"
5. UFSAR Section 8.3
6. NEI 99-01 SU1

Category: M – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

MA1.1 Alert

AC power capability, Table M-1, to Unit 1(2) 4160V emergency buses H and J 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 SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.

Table M-1 AC Power Sources
Offsite: Unit 1 • Transfer Bus D • Transfer Bus F • Station Bus 1B • Station Bus 2B Unit 2 • Transfer Bus E • Transfer Bus F • Station Bus 2C • Station Bus 1A **Onsite:** • 1(2)H EDG • 1(2)J EDG • AAC (SBO) Diesel Generator (if already aligned)

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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:

Table M-1 provides a list of offsite and onsite AC electrical power sources credited for this EAL.

Unit 1(2) 4160V emergency buses H and J are the emergency buses (ref. 1).

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC ~~SU1~~MU1.

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

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

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

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down, the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs). The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). Additional bus ties for Unit 1 exist between the 1H emergency bus to 1B station service bus and 1J emergency bus to 2B station service bus which can provide a second independent offsite power sources to each Unit 1 emergency bus. Unit 2 emergency busses can be cross tied between the following: 2C station service bus to 2H and 1A station service bus to 2J, which can provide a second independent offsite power source to each Unit 2 emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions. A bus that is powered from the SBO can be credited as being powered from an independent power source. However, since it takes longer than 15 minutes to align the SBO diesel generator, the generator must be "already aligned" to credit it as an AC power source.

Escalation of the emergency classification level would be via IC ~~SS1~~MS1.

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

Reference(s):

1. 11715-FE-1A, "Main One Line Diagram (Unit 1)"
2. 12050-FE-1A, "Main One Line Diagram (Unit 2)"
3. 1(2)-ECA-0.0, "Loss of All AC Power"
4. 0-AP-10, "Loss of Electrical Power"
5. UFSAR Section 8.3
6. NEI 99-01 SA1

Category: M – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

MS1.1 Site Area Emergency

Loss of **all** offsite and **all** onsite AC power to Unit 1(2) 4160V emergency buses H and J for ≥15 min. (Note 1)

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

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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:

For this EAL credit can be taken for any AC power source that has sufficient capability to operate equipment necessary to maintain a safe shutdown condition, such as FLEX generators, provided it can be aligned within the 15 minute classification criteria.

Unit 1(2) 4160V emergency buses H and J are the emergency buses (ref. 1).

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

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

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A

breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down, the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs). The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). Additional bus ties for Unit 1exist between the 1H emergency bus to 1B station service bus and 1J emergency bus to 2B station service bus which can provide a second independent offsite power sources to each Unit 1 emergency bus. Unit 2 emergency busses can be cross tied between the following: 2C station service bus to 2H and 1A station service bus to 2J, which can provide a second independent offsite power source to each Unit 2 emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions. A bus that is powered from the SBO can be credited as being powered from an independent power source.

Escalation of the emergency classification level would be via ICs ~~AG1~~RG1, FG1 or ~~SG1~~MG1.

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

Reference(s):

1. 11715-FE-1A, "Main One Line Diagram (Unit 1)"
2. 12050-FE-1A, "Main One Line Diagram (Unit 2)"
3. 1(2)-ECA-0.0, "Loss of All AC Power"
4. 0-AP-10, "Loss of Electrical Power"
5. UFSAR Section 8.3
6. NEI 99-01 SS1

Category: M –System Malfunction

Subcategory: 1 – Loss of Vital AC Power

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

EAL:

MG1.1 General Emergency
Loss of **all** offsite and **all** onsite AC power to Unit 1(2) 4160V emergency buses H and J
AND
Core Cooling-RED Path conditions met

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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:

For this EAL credit can be taken for any AC power source that has sufficient capability to operate equipment necessary to maintain a safe shutdown condition, such as the FLEX generators.

This IC addresses a prolonged loss of all power sources to AC emergency buses that results in degraded core cooling. 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 eventually 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 threshold is based on meeting either CSFST Core Cooling Red Path criteria (ref. 6, 7):

- Core Exit Thermocouple readings ≥1,200 °F.

- Core exit TCs are ≥700°F with RCS subcooling based on core exit TCs ≤25°F [75°F], no RCPs are running, and RVLIS full range is ≤48%

~~The~~ 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.

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

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

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

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down, the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs). The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). Additional bus ties for Unit 1 exist between the 1H emergency bus to 1B station service bus and 1J emergency bus to 2B station service bus which can provide a second independent offsite power sources to each Unit 1 emergency bus. Unit 2 emergency busses can be cross tied between the following: 2C station service bus to 2H and 1A station service bus to 2J, which can provide a second independent offsite power source to each Unit 2 emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions. A bus that is powered from the SBO can be credited as being powered from an independent power source.

Reference(s):

1. 11715-FE-1A, "Main One Line Diagram (Unit 1)"
2. 12050-FE-1A, "Main One Line Diagram (Unit 2)"
3. 1(2)-ECA-0.0, "Loss of All AC Power"

4. 0-AP-10, "Loss of Electrical Power"
5. UFSAR Section 8.3
6. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 2 Core Cooling"
7. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"
8. NEI 99-01 SG1

Category: M – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

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

EAL:

MS2.1	**Site Area Emergency**
Indicated voltage is < 105 VDC on **all** vital 125 VDC battery buses for ≥15 min. (Note 1)	

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

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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:

There are four independent 125 volt DC systems for each unit.

Each system consists of 125 volt DC distribution panels and its respective battery and a battery charger. The batteries 1(2)-I, 1(2)-II, 1(2)-III, and 1(2)-IV supply power only if the battery chargers fail or if the demand exceeds the capacity of the chargers. The batteries are rated for a minimum of two hours (ref. 1, 2).

A battery terminal voltage of 105 volts DC is the minimum voltage required to ensure proper operation of equipment connected to the DC bus (ref. 3).

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

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

Escalation of the emergency classification level would be via ICs ~~AG1~~RG1, FG1 or MG2~~SG8~~.

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

Reference(s):

1. 1(2)-AP-10, "Loss of Electrical Power"
2. UFSAR Section 8.3.2, "Direct Current Power System"
3. 0-OP-6.4, "Operation of the SBO Diesel (SBO Event)"
4. NEI 99-01 SS8

Category: M –System Malfunction

Subcategory: 2 – Loss of Vital DC Power

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

EAL:

MG2.1 General Emergency

Loss of **all** offsite and **all** onsite AC power to Unit 1(2) 4160V emergency buses H and J for ≥15 min. (Note 1)

AND

Indicated voltage is < 105 VDC on **all** vital 125 VDC battery buses for ≥15 min. (Note 1)

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

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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 IC addresses a concurrent and prolonged loss of both emergency AC and vital DC power. A loss of all emergency AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both emergency AC and vital DC power will lead to multiple challenges to fission product barriers.

For this EAL credit can be taken for any AC power source that has sufficient capability to operate equipment necessary to maintain a safe shutdown condition, such as the FLEX generators.

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A

breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down, the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs). The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). Additional bus ties for Unit 1exist between the 1H emergency bus to 1B station service bus and 1J emergency bus to 2B station service bus which can provide a second independent offsite power sources to each Unit 1 emergency bus. Unit 2 emergency busses can be cross tied between the following: 2C station service bus to 2H and 1A station service bus to 2J, which can provide a second independent offsite power source to each Unit 2 emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions. A bus that is powered from the SBO can be credited as being powered from an independent power source.

There are four independent 125 volt DC systems for each unit.

Each system consists of 125 volt DC distribution panels and its respective battery and a battery charger. The batteries 1(2)-I, 1(2)-II, 1(2)-III, and 1(2)-IV supply power only if the battery chargers fail or if the demand exceeds the capacity of the chargers. The batteries are rated for a minimum of two hours (ref. 4, 6).

A battery terminal voltage of 105 volts DC is the minimum voltage required to ensure proper operation of equipment connected to the DC bus (ref. 7).

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.

Reference(s):

1. 11715-FE-1A, “Main One Line Diagram (Unit 1)”
2. 12050-FE-1A, “Main One Line Diagram (Unit 2)”
3. 1(2)-ECA-0.0, “Loss of All AC Power”
4. 0-AP-10,” Loss of Electrical Power”
5. UFSAR Section 8.3
6. UFSAR Section 8.3.2, “Direct Current Power System”
7. 0-OP-6.4, “Operation of the SBO Diesel (SBO Event)”
8. NEI 99-01 SG8

Category: M – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

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

EAL:

MU3.1	**NOUE**
An UNPLANNED event results in the inability to monitor one or more Table M-2 parameters from within the Control Room for ≥15 min. (Note 1)	

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

Table M-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core exit TC temperature • Level in at least one SG • Auxiliary feedwater flow to at least one SG

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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:

Applicable safety system parameters are listed in Table M-2.

The Plant Computer System/Safety Parameter Display System (SPDS) serve as redundant indicators which may be utilized as compensatory measures in lieu of the Control Room indicators associated with safety functions (ref. 1, 2).

The Inadequate Core Cooling Monitor (ICCM) System consists of three redundant subsystems that provide continuous control room displays: Core Exit Thermocouple (CET) System, Core Cooling Monitor (CCM) System, and Reactor Vessel Level Instrumentation System (RVLIS) (ref. 3).

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

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

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

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

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

Escalation of the emergency classification level would be via IC MA3~~SA2~~.

Reference(s):

1. UFSAR Section 7.7.1.10, "Computer System"
2. UFSAR Section 7.8, "Emergency Response to Accidents"
3. UFSAR Section 7.9, "Inadequate Core Cooling Monitor (ICCM) System"

4. NEI 99-01 SU2

Category: M – 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:

MA3.1 Alert

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

AND

Any significant transient is in progress, Table M-3

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

Table M-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core exit TC temperature • Level in at least one SG • Auxiliary feedwater flow to at least one SG

Table M-3 Significant Transients
• Automatic turbine runback > 25% thermal reactor power • Electrical load rejection > 25% full electrical load • Reactor Trip • SI actuation

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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:

Applicable safety system parameters are listed in Table M-2.

Significant transients are listed in Table M-3.

The Plant Process Computer System/Safety Parameter Display System (SPDS) serve as redundant indicators which may be utilized as compensatory measures in lieu of the Control Room indicators associated with safety functions (ref. 1, 2).

The Inadequate Core Cooling Monitor (ICCM) System consists of three redundant subsystems that provide continuous control room displays: Core Exit Thermocouple (CET) System, Core Cooling Monitor (CCM) System, and Reactor Vessel Level Instrumentation System (RVLIS) (ref. 3).

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

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

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

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

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

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

Reference(s):

1. UFSAR Section 7.7.1.10, "Computer System "
2. UFSAR Section 7.8, "Emergency Response to Accidents"
3. UFSAR Section 7.9, "Inadequate Core Cooling Monitor (ICCM) System"
4. NEI 99-01 SA2

Category: M – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

MU4.1	**NOUE**
With letdown in service, Reactor Coolant Letdown Radiation Monitor 1(2)CH-RI-128(228) > 1.50E+04 mrem/hr	

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit (60 µCi/cc DEI-131) specified in Technical Specifications (ref. 1, 2). This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Per Engineering Calculation PA-0234, Rev. 1, the threshold value is indicative of more than 60 µCi/cc DEI-131 accident mix after 1 hour of decay. A monitor reading in excess of the threshold value 1.50E+04 mrem/hr (equivalent to 60 µCi/cc) indicates a challenge to the Technical Specification allowable limits for fuel clad degradation (ref. 1).

A portion of the letdown stream flows past radiation monitors 1(2)-CH-RI-128(228) to detect fission product activity in the reactor coolant and warn of a potential fuel element failure (ref. 3).

Escalation of the emergency classification level would be via IC~~s~~ FA1 or the ~~Recognition~~ Category ~~A~~ R ICs.

Reference(s):

1. Calculation No. PA-0234, Rev. 1, "Post Accident Letdown Radiation Monitor Response for North Anna"
2. Technical Specifications 3.4.16, "RCS Specific Activity"
3. UFSAR Section 11.4.2.15, "Reactor Coolant Letdown Gross Activity Monitors"
4. NEI 99-01 SU3

Category: M – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

MU4.2 NOUE

Dose rate at 1 ft. from an unpressurized RCS sample ≥Table M-4

Table M-4 Tech. Spec. Coolant Activity Dose Rates	
Time > Shutdown (hrs)	**mR/hr/ml**
≤2	0.70
> 2 – ≤8	0.50
> 8	0.30

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

Basis:

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

Per Engineering Calculation RA-0059 (ref. 1), dose rate is assumed to result from radioactive iodines (I-131 thru I-135) in RCS in concentrations corresponding to 60 µCi/gm DEI-131. This value corresponds to the Technical Specification coolant activity limit for iodine spike at full power operations (ref. 2). The values contained in Table M-4 (Tech. Spec. Coolant Activity Dose Rates) represent expected one foot dose rates per ml of sample based on time since reactor shutdown to the time when the sample is taken. The expected dose rate is a near linear relationship with the volume of the sample, so any volume collected can be determined by dividing the measured dose rate by the sample volume and comparing to the threshold value from Table M-4 for the applicable time frame. These dose rates assume no emergency core cooling system (ECCS) injection so there is no dilution credited which would vary coolant volume. Values in the table have been rounded for ease of use. The > 8 hour threshold is conservative up to 24 hours following reactor shutdown. After 24 hours, the expected response from radioactive iodine levels off. Therefore, the value shown for > 8 hours applies for all samples taken 8 hours or more since reactor shutdown.

The values specified in Table M-4 were developed using a method to minimize error (+/-) for the threshold value within each defined time period. Values were chosen to minimize error from the highest to lowest dose rate within each range.

It should be noted that this EALs is primarily directed toward mechanical damage to the clad not involving inadequate core cooling (ICC) sequences. Clad damage due to ICC sequences is addressed by the fuel clad and CTMT fission product barrier thresholds (Category F).

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

Reference(s):

1. RA-0059, "Detector Response to an RCS Sample for EAL Classification of Fuel Clad Degradation and Barrier Loss"
2. Technical Specifications 3.1.D
3. NEI 99-01 SU3

Category: M – System Malfunction

Subcategory: 4 – RCS Activity

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

EAL:

MU4.3	**NOUE**
Sample analysis indicates that a reactor coolant activity value is > an allowable limit specified in Technical Specification 3.4.16	

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

Basis:

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

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

Reference(s):

1. Technical Specifications 3.4.16, "RCS Specific Activity"
2. NEI 99-01 SU3

Category: M – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

MU5.1 NOUE

RCS unidentified or pressure boundary leakage > 10 gpm for ≥15 min.

OR

RCS identified leakage > 25 gpm for ≥15 min.

OR

Leakage from the RCS to a location outside containment > 25 gpm for ≥15 min.

(Note 1)

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

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

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

Basis:

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

Once the RCS leak rate has been quantified to be greater than the specified value, failure to isolate the leak within 15 minutes, or if known that the leak cannot be isolated within 15 minutes, from the time of leak rate quantification, requires immediate classification.

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

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

Unidentified leakage is all leakage (except RCP seal water injection or leak-off) that is not identified leakage. Pressure Boundary leakage is leakage (except SG leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall. Generally, leakage into *closed* systems, or leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the unidentified leakage monitoring systems or not to be from a fault in the reactor coolant pressure boundary, are called identified leakages.

The leak rate values for each condition ~~EAL~~ were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). The first condition ~~EAL #1~~ uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage (ref. 3, 4, 5).

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. ~~For PWRs, a~~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, locally or remotely). ~~For BWRs, a stuck-open Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.~~

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

Reference(s):

1. Technical Specification Section 1.1, "Definitions"
2. Technical Specification 3.4.13, "RCS Operational Leakage"
3. 1(2)-PT-52.2, "Reactor Coolant System Leak Rate (Hand Calculation)"
4. 1(2)-PT-52.2A, "Reactor Coolant System Leak Rate (Computer Calculation)"
5. 1(2)-AP-16, "Increasing Primary Plant Leakage"
6. NEI 99-01 SU4

Category: M – System Malfunction

Subcategory: 6 – RPS Failure

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

EAL:

MU6.1 NOUE

An automatic trip did **not** shut down the reactor as indicated by reactor power ≥5% after **any** RPS setpoint is exceeded

AND

A subsequent automatic trip **OR** manual trip (trip switches or manual turbine trip) are successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

Note 8: A manual trip 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

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~~ EAL addresses a failure of the RPS to initiate or complete an automatic ~~or manual~~ reactor ~~(trip [*PWR*] / scram [*BWR*])~~that results in a reactor shutdown (reactor power < 5%), and either a subsequent operator manual action taken at the reactor control consoles or an automatic ~~(trip[*PWR*] / scram [*BWR*])~~ (i.e., any subsequent RPS setpoint trip) is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor ~~(trip[*PWR*] / scram [*BWR*])~~, operators will promptly initiate manual actions at the reactor control consoles to shut down the reactor (e.g., initiate a manual reactor ~~(trip[*PWR*] / scram [*BWR*])~~ using the reactor trip switches or manually tripping the main turbine). 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 (< 5%).

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

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor ~~(trip~~

using the reactor trip switches or manually tripping the main turbine ~~[*PWR*] / scram [*BWR*]~~)). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at ~~back-panels or~~ other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles". ~~Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action. [*BWR*]~~

A reactor trip resulting from actuation of the ATWS Mitigation System Actuation Circuitry (AMSAC) logic is considered a successful subsequent automatic reactor trip for the purposes of this EAL (ref. 3).

The plant response to the failure of an automatic ~~or manual reactor (~~trip~~[PWR] / scram [BWR])~~ will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC ~~SA5~~MA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC ~~SA5~~ MA6 or FA1, an ~~Unusual Event~~NOUE declaration is appropriate for this event.

A reactor shutdown is determined ~~in accordance with applicable Emergency Operating Procedure criteria~~consistent with CSFST Subcriticality Red path criteria (ref. 1). Because the power level threshold for subcriticality RED path (5%) is the same as the Power Operation operating mode transition power, this EAL is only applicable in Mode 1.

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

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

In the event that the operator identifies a reactor trip is IMMINENT and initiates a successful manual reactor trip before the automatic RPS trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. However, if subsequent manual reactor trip actions fail to shut down the reactor, the event escalates to the Alert under EAL MA6.1.

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 1 Subcriticality"
2. 1(2)-E-0, "Reactor Trip or Safety Injection"
3. UFSAR Section 7.2.1.1.6, "Turbine Trip-Reactor Trip"
4. NEI 99-01 SU5

Category: M – System Malfunction

Subcategory: 6 – RPS Failure

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

EAL:

MU6.2 NOUE

A manual trip did **not** shut down the reactor as indicated by reactor power ≥5%

AND

A subsequent manual trip (trip switches or manual turbine trip) **OR** automatic trip is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

Note 8: A manual trip 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

Definition(s):

None

Basis:

This ~~IC~~ EAL addresses a failure of ~~the RPS to initiate or complete an automatic or~~a manual reactor ~~(trip [*PWR*] / scram [*BWR*])~~ that results in a reactor shutdown (reactor power < 5%), and either a subsequent operator manual action taken at the reactor control consoles or an automatic ~~(trip[*PWR*] / scram [*BWR*])~~ is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

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

~~If an initial manual reactor (trip [*PWR*] / scram [*BWR*]) is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor (trip[*PWR*] / scram [*BWR*])) using a different switch).~~

Depending upon several factors, the initial or subsequent effort to manually ~~(trip [*PWR*] / scram [*BWR*])~~the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor ~~(trip [*PWR*] / scram [*BWR*])~~signal. If a subsequent manual or automatic ~~(trip [*PWR*] / scram [*BWR*])~~is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems (< 5%) (ref. 1, 2).

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 ~~(~~trip using the reactor trip switches or manually tripping the main turbine ~~[*PWR*] / scram [*BWR*])~~). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at ~~back panels or other~~ locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

~~Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action. [BWR]~~

A reactor trip resulting from actuation of the ATWS Mitigation System Actuation Circuitry (AMSAC) logic is considered a successful subsequent automatic reactor trip for the purposes of this EAL (ref. 3).

The plant response to the failure of ~~an automatic or~~ manual reactor ~~(~~trip ~~[PWR] / scram [BWR])~~ will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC ~~SA5~~MA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC ~~SA5~~ MA6 or FA1, an ~~Unusual Event~~NOUE declaration is appropriate for this event.

A reactor shutdown is determined consistent with CSFST Subcriticality Red path criteria (ref. 1).~~in accordance with applicable Emergency Operating Procedure criteria.~~ Because the power level threshold for subcriticality RED path (5%) is the same as the Power Operation operating mode transition power, this EAL is only applicable in Mode 1.

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

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

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 1 Subcriticality"
2. 1(2)-E-0, "Reactor Trip or Safety Injection"
3. UFSAR Section 7.2.1.1.6, "Turbine Trip-Reactor Trip"
4. NEI 99-01 SU5

Category: M – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Automatic or manual trip 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:

MA6.1 Alert

An automatic or manual trip did **not** shut down the reactor as indicated by reactor power ≥5%

AND

Subsequent automatic or manual trip actions (trip switches and manual turbine trip) are **not** successful in shutting down the reactor as indicated by reactor power ≥5% (Note 8)

Note 8: A manual trip 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

Definition(s):

None

Basis:

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

A manual action at the reactor control 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 ~~(~~trip~~[PWR] / scram [BWR])~~ using the reactor trip switches or manually tripping the main turbine). This action does not include locally tripping reactor trip and bypass breakers, 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 opening breakers). Actions taken at ~~back panels or~~ other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

~~Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action. [BWR]~~

A reactor trip resulting from actuation of the ATWS Mitigation System Actuation Circuitry (AMSAC) logic is considered a successful subsequent automatic reactor trip for the purposes of this EAL (ref. 3). Therefore an Alert classification would not be required.

The plant response to the failure of an automatic or manual reactor ~~(~~trip ~~[PWR] / scram [BWR])~~ will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shut_down the reactor is prolonged enough to cause a challenge to the core cooling ~~[PWR] / RPV water level [BWR]~~or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC ~~S~~MS6~~5~~. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC ~~S~~MS6~~5~~ or FS1, an Alert declaration is appropriate for this event.

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

A reactor shutdown is determined consistent with CSFST Subcriticality Red path criteria (ref. 1).~~in accordance with applicable Emergency Operating Procedure criteria.~~ Because the power level threshold for subcriticality RED path (5%) is the same as the Power Operation operating mode transition power, this EAL is only applicable in Mode 1.

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 1 Subcriticality"
2. 1(2)-E-0, "Reactor Trip or Safety Injection"
3. UFSAR Section 7.2.1.1.6, "Turbine Trip-Reactor Trip"
4. NEI 99-01 SA5

Category: M – System Malfunction

Subcategory: 2 – RPS Failure

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

EAL:

MS6.1 Site Area Emergency

An automatic or manual trip did **not** shut down the reactor as indicated by reactor power ≥5%

AND

All actions taken to shut down the reactor are **not** successful as indicated by reactor power ≥5%

AND EITHER:

- Core Cooling-RED Path conditions met
- Heat Sink-RED Path conditions met

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

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

Reactor shutdown achieved by use of other trip actions such as locally opening supply breakers, emergency boration, or manually driving control rods are also credited as a successful manual trip if reactor power is < 5% before indications of an extreme challenge to either core cooling or heat removal exist (ref. 1, 2, 3).

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

A reactor shutdown is determined consistent with CSFST Subcriticality Red path criteria (ref. 1). ~~in accordance with applicable Emergency Operating Procedure criteria~~. Because the power level threshold for subcriticality RED path (5%) is the same as the Power Operation operating mode transition power, this EAL is only applicable in mode 1.

A severe challenge to adequate core cooling is based on meeting the Core Cooling Red path criteria (ref. 4, 5):

- Core Exit Thermocouple readings ≥1,200 °F.
- Core exit TCs are ≥700°F with RCS subcooling based on core exit TCs ≤25°F [75°F], no RCPs are running, and RVLIS full range is ≤48%.

The severe challenge to RCS heat removal is based on meeting the Heat Sink Red path criteria of both of the following conditions existing (ref. 6, 7):

- Narrow Range levels in all SGs < 11% [22%]
- Total feedwater flow to SGs ≤340 gpm

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

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 1 Subcriticality"
2. 1(2)-FR-S.1, "Response to Nuclear Power Generation / ATWS"
3. 1(2)-E-0, "Reactor Trip or Safety Injection"
4. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 2 Core Cooling"
5. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"
6. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 3 Heat Sink"
7. 1(2)-FR-H.1, "Response to Loss of Secondary Heat Sink"
8. NEI 99-01 SS5

Category: M – System Malfunction

Subcategory: 7 – Loss of Communications

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

EAL:

MU7.1 NOUE

Loss of **all** Table M-5 onsite communication methods

OR

Loss of **all** Table M-5 State and local agency communication methods

OR

Loss of **all** Table M-5 NRC communication methods

Table M-5 Communication Methods

System	Onsite	State/ Local	NRC
Radio Communications System	X		
Public Address and Intercom System	X		
Private Branch Telephone Exchange (PBX)	X	X	X
Sound Powered Telephone System	X		
Commercial Telephone System		X	X
Automatic Ring Downs (SONET Ring)		X	
Instaphone Loop		X	
Dedicated NRC Communications			X

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

Basis:

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

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

The first EAL condition ~~EAL #1~~ addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition ~~EAL #2~~ addresses a total loss of the communications methods used to notify all ~~OROs~~ State and local agencies of an emergency declaration. The ~~OROs~~ State and local agencies referred to here are the Commonwealth of Virginia and local communities. ~~(see Developer Notes)~~

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

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

Reference(s):

1. North Anna Power Station Emergency Plan, Section 7.2, “Communications Systems”
2. UFSAR Section 7.7.1
3. NEI 99-01 SU6

Category: M – System Malfunction

Subcategory: 8 – Containment Failure

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

EAL:

MU8.1 NOUE

Any penetration is **not** closed within 15 min. of a VALID Phase A or B isolation signal

OR

CTMT pressure > 28 psia with < one full train of CTMT depressurization equipment (Note 11) operating per design for ≥15 min.

(Note 1)

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

Note 11: One full train of containment depressurization equipment consist of one Quench Spray (QS) System and one Recirculation Spray (RS) System from either train operating together.

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown

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 ~~IC~~ EAL addresses a failure of one or more containment penetrations to automatically isolate (close) when required by an actuation signal (ref. 1). It also addresses an event that results in high containment pressure with a concurrent failure of containment pressure control systems. Absent challenges to another fission product barrier, either condition represents potential degradation of the level of safety of the plant.

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

~~EAL #2~~The second condition addresses a condition where containment pressure is greater than the setpoint (28 psia) at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design (ref. 4, 5).

The Quench Spray (QS) System, operating in conjunction with the Recirculation Spray (RS) System, is designed to cool and depressurize the containment structure to less than 2.0 psig in one hour and sub-atmospheric pressure in less than 6 hours following a Design Basis Accident. The combination of required equipment can be obtained from using equipment on either emergency busses in order to meet the "one full train" requirement (ref. 2, 3).

The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment sprays~~or ice condenser fans~~) are either lost or performing in a degraded manner.

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

Reference(s):

1. UFSAR Section 6.2.4, "Containment Isolation System"
2. Technical Specifications Section B 3.6.6, "Quench Spray (QS) System"
3. Technical Specifications Section B 3.6.7, "Recirculation Spray (RS) System"
4. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 5 Containment"
5. 1(2)-FR-Z.1, "Response to High Containment Pressure"
6. NEI 99-01 SU7

Category: M – System Malfunction

Subcategory: 9 – Hazardous Event Affecting Safety Systems

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

EAL:

MA9.1 Alert

The occurrence of **any** Table M-6 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 second train of the SAFETY SYSTEM needed for the current operating mode
- Event damage has resulted in VISIBLE DAMAGE to the second 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 M-6 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/SEM

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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.

Basis:

This IC 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 second 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 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.

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

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

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

An event affecting equipment common to two or more trains of a safety system (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the common equipment) should be classified as an Alert under this EAL, as appropriate to the plant mode. By affecting the functionality of multiple trains of a safety system, the loss of the common equipment effectively meets the two-train impact criteria that underlie the EALs and bases.

An event affecting a single-train safety system (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under this EAL because the two-train impact criteria that underlie the EALs and bases would not be met. If an event affects a single-train safety system, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on SEM judgement.

An event that affects two trains of a safety system (e.g., one train has indications of degraded performance and the other VISIBLE DAMAGE) that also has one or more additional trains should be classified as an Alert under this EAL, as appropriate to the plant mode. This approach maintains consistency with the two-train impact criteria that underlie the EALs and bases and is warranted because the event was severe enough to affect the functionality of two trains of a safety system despite plant design criteria associated with system and system train separation and protection. Such an event may have caused other plant impacts that are not immediately apparent.

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

This hot condition EAL is equivalent of the cold condition EAL CA6.1.

Reference(s):

1. EP FAQ 2016-002
2. NEI 99-01 SA9

Background

NEI 99-01, Rev. 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:

> *The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.*
>
> *The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).*

Further, as specified in IC HA5:

> *The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.*

NAPS Table R-2 and H-2 Bases

A review of station operating procedures identified the following mode dependent in-plant actions and associated areas that are required for normal plant operation, cooldown or shutdown:

In-Plant Actions (NAPS)	Safe Shutdown Area	Modes
Chemistry to perform RCS isotopic analysis	AB El 274	1, 2
Ensure boron concentration for Cold Shutdown	AB El 274	3, 4
Sample RCS to place RHR in service	AB El 274	3, 4
I&C to perform PT-44.41	Instrument Rack Room	4
Place RHR in service per OP-14.1	AB EL 274' Cable Vault & Tunnels	4

Control Room ventilation systems have adequate engineered safety/design features in place to preclude a Control Room evacuation due to the external release of a hazardous gas (UFSAR Section 9.4.1 Main Control Room and Relay Rooms). Therefore, the Control Room is not included in this assessment or in Tables R-2/H-2.

Ref: OP-3.7, "Unit Shutdown from Mode 1 to Mode 5 for Refueling"

Table R-2 & H-2 Results

Table R-2/H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode**
Aux. Building El 274'	1, 2, 3, 4
Instrument Rack Rooms	4
Cable Vault & Tunnels	

ATTACHMENT 3

NAPS EAL TECHNICAL BASES DOCUMENT (Final)

Virginia Electric and Power Company
(Dominion Energy Virginia)
North Anna Power Station Units 1 and 2 and ISFSIs

Emergency Action Level Technical Bases Document
North Anna Power Station

(Final)

Table of Contents

1.0 INTRODUCTION

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the NEI 99-01, Rev. 6, EAL Upgrade Project for North Anna Power Station (NAPS). It should be used to facilitate review of the NAPS EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EPIP-1.01, Emergency Manager Controlling Procedure, may use this document as a technical reference in support of EAL interpretation. This information may assist the Station Emergency Manager (SEM) in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training and for explaining event classifications to off-site officials.

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

Since the information in a basis document can affect emergency classification decision-making (e.g., the SEM refers to it during an event), the NRC staff expects that changes to the basis document will be evaluated in accordance with the provisions of 10 CFR 50.54(q). For Dominion Energy sites, a 10 CFR 50.54(q)(3) screening/evaluation will be performed to evaluate changes to this document.

Dominion Energy fleet procedure CM-AA-400, "10 CFR 50.59 and 10 CFR 72.48 – Changes, Tests and Experiments," provides a method to determine the impacts to licensing basis documents when changes are proposed to procedures, including changes to Abnormal Operating Procedures (AOPs) and Emergency Operating Procedures (EOPs). The 50.59/72.48 applicability review form specifically requires that the effect of a proposed procedure change on the Emergency Plan (and associated EALs) be reviewed/assessed. When impacts to the Emergency Plan are identified, a separate review in accordance to 10 CFR 50.54(q) will be performed to determine the acceptability of the proposed procedure change.

2.0 DISCUSSION

2.1 Background

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the North Anna Power Station (NAPS) Emergency Plan.

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

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

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

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

Subsequently, Rev. 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, Rev. 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 (ref. 4.1.1), NAPS conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

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

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

The primary fission product barriers are:

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

B. Reactor Coolant System Barrier (RCS): The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

C. Containment Barrier (CTMT): The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the Emergency Classification Level (ECL) from an Alert to a Site Area Emergency or a General Emergency.

2.3 Fission Product Barrier Classification Criteria

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

Alert:

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

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

2.4 EAL Organization

The NAPS EAL scheme includes the following features:

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

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

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

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

The EALs are pre-determined, site-specific, observable thresholds for determining whether an Initiating Condition (IC) has occurred and that an EAL threshold was met or exceeded. Thus failure to evaluate the IC and EAL together could result in an incorrect declaration.

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:	
R – Abnormal **R**ad Levels / Rad Effluent	1 – Radiological Effluent 2 – Irradiated Fuel Event 3 – Area Radiation Levels
H – **H**azards and Other Conditions Affecting Plant Safety	1 – Security 2 – Seismic Event 3 – Natural or Technological Hazard 4 – Fire 5 – Hazardous Gas 6 – Control Room Evacuation 7 – SEM Judgment
E – Independent Spent Fuel Storage Installation (ISFSI)	1 – Confinement Boundary
Hot Conditions:	
M – System **M**alfunction	1 – Loss of Emergency 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 – Containment Failure 9 – Hazardous Event Affecting Safety Systems
F – **F**ission Product Barrier Degradation	None
Cold Conditions:	
C – **C**old Shutdown / Refueling System Malfunction	1 – RCS Level 2 – Loss of Emergency AC Power 3 – RCS Temperature 4 – Loss of Vital DC Power 5 – Loss of Communications 6 – Hazardous Event Affecting Safety Systems

2.5 Technical Bases Information

EAL technical bases are provided in Attachment 1 for each EAL according to EAL group (Any, Hot, Cold), EAL category (R, C, E, F, H and M) and EAL subcategory. A summary is given at the beginning of each group, which provides a brief description of the category.

For each EAL, the following information is provided:

Category Letter & Title

Subcategory Number & Title

Initiating Condition (IC)

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

EAL iIdentifier (enclosed in rectangle)

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

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

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

Classification (enclosed in rectangle):

General Emergency (G), Site Area Emergency (S), Alert (A) or NOUE (U).

EAL Wording (enclosed in rectangle)

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

Mode Applicability

One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown, 5 - Cold Shutdown, 6 - Refueling, D - Defueled, All - All mode (See Section 2.6 for operating mode definitions).

Notes (as applicable)

<u>Definitions:</u>

If the EAL wording contains a defined term, the definition of the term is included in this section. These definitions can also be found in Section 5.1.

<u>Basis:</u>

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

<u>Reference(s):</u>

Source documentation from which the EAL is derived.

2.6 Operational Mode Applicability

Technical Specifications, definition 1.C, assigns the following reactor operating modes for Power Operation through Refueling:

1 <u>Power Operation</u>

K_{eff} $\geq$0.99 and rated thermal power > 5%

2 <u>Startup</u>

K_{eff} $\geq$0.99 and rated thermal power $\leq$5%

3 <u>Hot Standby</u>

K_{eff} < 0.99 and average reactor coolant temperature T_{avg} $\geq$350°F

4 <u>Hot Shutdown</u>

K_{eff} < 0.99 and average reactor coolant temperature 350°F > T_{avg} > 200°F with all reactor vessel head closure bolts fully tensioned

5 <u>Cold Shutdown</u>

K_{eff} < 0.99 and average reactor coolant temperature T_{avg} $\leq$200°F with all reactor vessel head closure bolts fully tensioned

6 <u>Refueling</u>

One or more reactor vessel head closure bolts less than fully tensioned

D <u>Defueled</u>

All fuel assemblies have been removed from Containment

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

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the SEM must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the EAL plus the associated Operational 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.

3.1.1 Classification Timeliness

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, "Interim Staff Guidance, Emergency Planning for Nuclear Power Plants" (ref. 4.1.8).

3.1.2 Valid Indications

All emergency classification assessments shall be based upon valid indications, reports or conditions. A valid indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy.

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

3.1.3 Imminent Conditions

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

3.1.4 Planned vs. Unplanned Events

A planned work activity that results in an expected event or condition which meets or exceeds an EAL does not warrant an emergency declaration provided that: 1) the activity proceeds as planned, and 2) the plant remains within the limits imposed by the operating license. Such activities include planned work to test, manipulate, repair, maintain or modify a system or component. In these cases, the controls associated with the planning, preparation and execution of the work will ensure that compliance is maintained with all aspects of the operating license provided that the activity proceeds and concludes as expected. Events or conditions of this type may be subject to the reporting requirements of 10CFR 50.72 (ref. 4.1.4).

3.1.5 Classification Based on Analysis

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

3.1.6 SEM Judgment

While the EALs have been developed to address a full spectrum of possible events and conditions which may warrant emergency classification, a provision for classification based on operator/management experience and judgment is still necessary. The NEI 99-01 EAL scheme provides the SEM 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 SEM will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated in the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

3.2 Classification Methodology

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

When assessing an EAL that specifies a time duration for the potentially classifiable condition, the “clock” for the EAL time duration runs concurrently with the emergency classification process “clock.” For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01 (ref. 4.1.8).

3.2.1 Classification of Multiple Events and Conditions

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

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

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

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

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

3.2.2 Consideration of Mode Changes During Classification

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

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

3.2.3 Classification of Imminent Conditions

Although EALs provide specific thresholds, the SEM 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 SEM, meeting an EAL is IMMINENT, the emergency classification should be made as if the EAL has been met. While applicable to all emergency classification levels, this approach is particularly important at the higher emergency classification levels since it provides additional time for implementation of protective measures.

3.2.4 Emergency Classification Level Upgrading and Downgrading

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

As noted above, guidance concerning classification of rapidly escalating events or conditions is provided in RIS 2007-02 (ref. 4.1.2).

3.2.5 Classification of Short-Lived Events

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

3.2.6 Classification of Transient Conditions

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

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.

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

> An ATWS occurs and the high pressure ECCS systems fail to automatically start. The plant enters an inadequate core cooling condition (a potential loss of both the Fuel Clad and RCS Barriers). If an operator manually starts a high pressure ECCS system in accordance with an EOP step and clears the inadequate core cooling condition prior to an emergency declaration, then the classification should be based on the ATWS only.

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

3.2.7 After-the-Fact Discovery of an Emergency Event or Condition

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

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 (ref. 4.1.3) is applicable. Specifically, the event should be reported to the NRC in accordance with 10CFR 50.72 (ref. 4.1.4) within one hour of the discovery of the undeclared event or condition. The licensee should also notify appropriate State and local agencies in accordance with the agreed upon arrangements.

3.2.8 Retraction of an Emergency Declaration

Guidance on the retraction of an emergency declaration reported to the NRC is discussed in NUREG-1022 (ref. 4.1.3).

4.0 REFERENCES

4.1 Developmental

4.1.1 NEI 99-01, Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," (ADAMS Accession No. ML12326A805)

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

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

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

4.1.5 10 CFR 50.73, "Licensee Event Report System"

4.1.6 Technical Specifications for North Anna Units 1 and 2

4.1.7 VPAP-2103N, "Offsite Dose Calculation Manual (North Anna)"

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

4.1.9 NAPS UFSAR Section 2.1.1.3 "Boundaries for Establishing Effluent Release Limits"

4.1.10 North Anna Power Station ISFSI NRC Certificate of Compliance 1030 Amendment 1, Technical Specifications and SER

4.1.11 OU-AA-200, "Shutdown Risk Management"

4.1.12 SY-AA-101, "Security and Access Control"

4.1.13 NAPS UFSAR Section 9.1.4.3, "Fuel-Handling Structures"

4.1.14 RIS 2003-18, "Use of NEI 99-01 Methodology for Development of Emergency Action Levels" and related Supplements 1 and 2"

4.2 Implementing

4.2.1 EPIP-1.01, Emergency Manager Controlling Procedure

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

4.2.3 NAPS EAL Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

Selected terms used in Initiating Condition, EAL statements and EAL bases are set in all capital letters (e.g., ALL CAPS). These are defined terms that have specific meanings as used in this document. The definitions of these terms are provided below.

ALERT

Events are in progress, or have occurred, which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. **Any** releases are expected to be limited to small fractions of the EPA 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 NAPS ISFSI, Confinement Boundary is defined as the Sealed Surface Storage Cask (SSSC) or NUHOMS Dry Storage Canister (DSC) (ref. 4.1.10).

CONTAINMENT CLOSURE

The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions (ref. 4.1.11).

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:

- Notification of Unusual Event (NOUE)
- 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.

FAULTED

The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.

FIRE

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

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 NAPS 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 NAPS. 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.

NOTIFICATION of UNUSUAL EVENT

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.

OWNER CONTROLLED AREA (OCA)

The entire area contiguous to the PLANT PROTECTED AREA, owned by the Company and designated to be controlled for security reasons (ref. 4.1.12).

PLANT PROTECTED AREA

An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force (ref. 4.1.12).

PROJECTILE

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

REFUELING PATHWAY

Refueling cavity, fuel transfer canal, and spent fuel pit (SFP), but **not** including the reactor vessel, comprise the refueling pathway (ref. 4.1.13).

RUPTURED

The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

SAFETY SYSTEM

A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (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

The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment (ref. 4.1.9).

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.

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.

5.2 Abbreviations/Acronyms

°F .... Degrees Fahrenheit
° .... Degrees
µCi .... Micro Curie
AC .... Alternating Current
AFW .... Auxiliary Feedwater
AP .... Abnormal Procedure
ARM .... Area Radiation Monitor
ATWS .... Anticipated Transient Without Scram
CDE .... Committed Dose Equivalent
CET .... Core Exit Thermocouple
CFR .... Code of Federal Regulations
CPM .... Counts Per Minute
CR .... Control Room
CSFST .... Critical Safety Function Status Tree
CTMT .... Containment
DBA .... Design Basis Accident
DEF .... Defueled
DC .... Direct Current
DE .... Dose Equivalent
DEI-131 .... Dose Equivalent I-131
D/G .... Diesel Generator
DSC .... Dry Storage Canister
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
EDG .... Emergency Diesel Generator
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
FAA .... Federal Aviation Administration
FBI .... Federal Bureau of Investigation
FC .... Fuel Clad Barrier
FEMA .... Federal Emergency Management Agency
GE .... General Emergency
GPM .... Gallons Per Minute
Hr. .... Hour
IC .... Initiating Condition

ISFSI ....Independent Spent Fuel Storage Installation
K_{eff} .... Effective Neutron Multiplication Factor
LCO.......... Limiting Condition of Operation
LOCA ....Loss of Coolant Accident
LRW ....Liquid Radwaste
LWR ....Light Water Reactor
MCB ....Main Control Board
Min. ....Minute
MPH ....Miles Per Hour
mR, mRem, mrem, mREM ....milli-Roentgen Equivalent Man
MW..........Megawatt
NEI .... Nuclear Energy Institute
NPP..........Nuclear Power Plant
NRC .... Nuclear Regulatory Commission
NSSS ....Nuclear Steam Supply System
NORAD ....North American Aerospace Defense Command
NOUE.......... Notification of Unusual Event
OBE ....Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM .... Off-site Dose Calculation Manual
PAG .... Protective Action Guideline
PSIG .... Pounds per Square Inch Gauge
R ....Roentgen
RCS ....Reactor Coolant System
Rem, rem, REM .... Roentgen Equivalent Man
RPS..........Reactor Protection System
RVLIS..........Reactor Vessel Level Instrumentation System
SBO .... Station Blackout
SCBA .... Self-Contained Breathing Apparatus
SEM .... Station Emergency Manager
SSSC .... Sealed Surface Storage Cask
SFP .... Spent Fuel Pool (Pit)
SG..........Steam Generator
SI..........Safety Injection
SM.......... Shift Manager
SPDS ....Safety Parameter Display System
SRO .... Senior Reactor Operator
TC (T/C) .... Thermocouple
TEDE ....Total Effective Dose Equivalent

TAF .... Top of Active Fuel
TS .... Technical Specifications
TSC .... Technical Support Center
(U)FSAR .... (Updated) Final Safety Analysis Report
USGS .... United States Geological Survey

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

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

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

NAPS	**NEI 99-01, Rev. 6**	
EAL	**IC**	**Example EAL**
RG2.1	AG2	1
CU1.1	CU1	1
CU1.2	CU1	2
CU2.1	CU2	1
CU3.1	CU3	1
CU3.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1, 2, 3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA3.1	CA3	1, 2
CA6.1	CA6	1
CS1.1	CS1	1
CS1.2	CS1	2
CS1.3	CS1	3
CG1.1	CG1	1
CG1.2	CG1	2
EU1.1	EU1	1
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1, 2, 3
HU2.1	HU2	1

NAPS	NEI 99-01, Rev. 6	
EAL	IC	Example EAL
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
HG7.1	HG7	1
MU1.1	SU1	1
MU3.1	SU2	1
MU4.1	SU3	1
MU4.2	SU3	1
MU4.3	SU3	2
MU5.1	SU4	1, 2, 3
MU6.1	SU5	1

NAPS	NEI 99-01, Rev. 6	
EAL	**IC**	**Example EAL**
MU6.2	SU5	2
MU7.1	SU6	1, 2, 3
MU8.1	SU7	1, 2
MA1.1	SA1	1
MA3.1	SA2	1
MA6.1	SA5	1
MA9.1	SA9	1
MS1.1	SS1	1
MS2.1	SS8	1
MS6.1	SS5	1
MG1.1	SG1	1
MG2.1	SG8	1

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases

Category R – Abnormal Rad Release / Rad Effluent

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

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

At lower levels, abnormal radioactivity releases may be indicative of a failure of containment systems or precursors to more significant releases. At higher release rates, offsite radiological conditions may result which require offsite protective actions. Elevated area radiation levels in 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 required to safely operate and shutdown the plant also warrant emergency classification.

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1a – Radiological Effluent

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

EAL:

RU1.1 NOUE

Reading on SW-RM-130(230) CW Discharge Tunnel radiation monitor > 2 x the "Hi-Hi" setpoint for ≥60 min.
(Notes 1, 2, 3)

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

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

Note 3: If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

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

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

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

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

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

This EAL addresses normally occurring continuous radioactivity releases from monitored liquid effluent pathways (ref. 1).

In order to optimally be able to read the "2 times the Hi-Hi alarm setpoint" threshold, the range selector switch for the monitor should be in the "wide" position. Note: This is the normal position for the switch (ref. 2).

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

Reference(s):

1. VPAP-2103N, "Offsite Dose Calculation Manual (North Anna)"
2. 1(2)-ICP-SW-RM-130(230), "Discharge Tunnel Effluent Radiation Monitor (RM-SW-( )30) Calibration"
3. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1a – Radiological Effluent

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

EAL:

RU1.2 NOUE

Sample analysis for a liquid release indicates a concentration or release rate > 2 x the allocated ODCM limits for ≥60 min.
(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

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 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 liquid radiological release, monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.

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

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

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

This EAL addresses uncontrolled 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 lake/reservoir water systems, etc.).

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

Reference(s):

1. VPAP-2103N, "Offsite Dose Calculation Manual (North Anna)"
2. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1b – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1 mrem TEDE

EAL:

RU1.3 NOUE

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

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

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

Note 3: If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

Table R-1 Gaseous Effluent Monitor Classification Thresholds

Release Point & Monitor	GE	SAE	Alert	NOUE
Vent Stack A VG-RI-179-1 or 2	2.6E+08 µCi/sec	2.6E+07 µCi/sec	2.6E+06 µCi/sec	2.6E+05 µCi/sec
Vent Stack B VG-RI-180-1 or 2	2.0E+08 µCi/sec	2.0E+07 µCi/sec	2.0E+06 µCi/sec	2.0E+05 µCi/sec
Process Vent GW-RI-178-1 or 2	3.5E+08 µCi/sec	3.5E+07 µCi/sec	3.5E+06 µCi/sec	3.5E+05 µCi/sec
Main Steam Line MS-RI-170 (270) MS-RI-171 (271) MS-RI-172 (272)	1.3E+03 mR/hr	1.3E+02 mR/hr	1.3E+01 mR/hr	N/A
TD AFW Pump EXH MS-RI-176 (276)	6.0E+01 mR/hr	6.0E +00 mR/hr	6.0E-01 mR/hr	N/A

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 IC addresses a potential decrease in the level of safety of the plant as indicated by a low-level radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release). It includes any gaseous radiological release, monitored or un-monitored.

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 effluent pathways (ref. 1, 2).

The basis for the NOUE values correspond to any unplanned release of gaseous effluent radioactivity to the environment that will result in greater than 1 mrem TEDE for 60 minutes or longer. This NOUE gaseous release criterion is being used consistently across all operating nuclear units at Dominion Energy. The reason this alternative criterion is required is due to the fact that for some effluent gaseous release pathways, the resulting calculated NOUE threshold following the NEI 99-01 guidance of two times the site-specific effluent release limit would result in a NOUE threshold value greater than the corresponding calculated ALERT threshold based on exceeding 10 mrem TEDE. For the other gaseous release pathways that did not show an incongruent relationship when compared to the ALERT threshold, many showed NOUE values essentially equivalent to 1 mrem TEDE when applying the guidance in NEI 99-01 of a value set at two times the site specific effluent release limit. The fact that, (1) many of the gaseous release pathway NOUE values following NEI 99-01 guidance were essentially equivalent to 1 mrem TEDE, (2) application of an alternative definition set at a value of 1 mrem TEDE results in a more limiting value for those release paths that showed incongruent comparison to the corresponding ALERT threshold, and (3) NOUE criterion set at a value ten (10) times lower than the ALERT threshold provides a logical and consistent escalation between each classification level, provides justification for the NOUE criterion of 1 mrem TEDE. This single Initiating Condition (IC) definition for gaseous releases at the NOUE level is being applied to maintain consistency across the Dominion Energy nuclear fleet and to reduce confusion and human error potential if two different (IC) definitions were applied (ref. 2).

Classification thresholds within Table R-1 were generated using the MIDAS dose assessment code. Inputs to MIDAS use most prevalent meteorological data and expected release point parameters. An assumed one-hour decay since shutdown and a one-hour release duration are applied. Mitigating reduction mechanisms (e.g., decay, sprays, filters) input into MIDAS for each accident type determined the radiological release source term consistent with the guidance provided in NUREG-1228.

The MGPI radiation monitors for 1-GW-RI-178-1 & 2, 1-VG-RI-179-1 & 2 and 1-VG-RI-180-1 & 2 consist of a "normal" (or low) and an "accident" (or high) range device. The "normal" range radiation monitor flowpath is isolated at a predetermined value at which time the "accident" range radiation monitor is automatically aligned for operation. The "normal" range radiation monitor must be manually put back in service when flowpath activity trends down.

Due to the fact that there are no ODCM limits on steam safeties or auxiliary feedwater exhausts and the limited ability for these respective radiation monitors to detect low level radioactivity in these steam line configurations, the NOUE classification thresholds for the steam safeties and auxiliary feedwater exhaust are being labeled N/A (not applicable)

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

Reference(s):

1. VPAP-2103N, "Offsite Dose Calculation Manual (North Anna)"
2. RP 08-22, "North Anna Abnormal Rad Release Gaseous EAL Thresholds based on NEI 99-01, Rev. 6"
3. DC NA-11-01082, "Main Steam Radiation Monitor Replacement"
4. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1b – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1 mrem TEDE

EAL:

RU1.4	**NOUE**

Sample analysis for a gaseous release indicates a concentration or release rate > 2 x the allocated ODCM limits for ≥60 min. (Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

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

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

This EAL addresses uncontrolled gaseous releases that are detected by sample analyses or environmental surveys.

Calculation RP 08-22 (ref. 2) demonstrates how a release rate limit based on 2 x the allocated REMODCM limit will produce essentially 1 mrem TEDE assuming most prevalent meteorological dispersion.

Most prevalent meteorology represents conditions that would most likely to exist (based on most prevalent stability class and average wind speed within that stability class). Dispersion based on most prevalent meteorology differs from that assumed in the REMODCM which uses annual average meteorology. Dispersion based on actual meteorological conditions at the time of the emergency (most prevalent) can be 10 – 20 times higher than the annual average dispersion prescribed for use in an ODCM.

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

Reference(s):

1. VPAP-2103N, "Offsite Dose Calculation Manual (North Anna)"
2. RP 08-22, "North Anna Abnormal Rad Release Gaseous EAL Thresholds based on NEI 99-01", Rev. 6
3. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.1 Alert

Reading on **any** Table R-1 effluent radiation monitor > column "ALERT" for ≥15 min. (Notes 1, 2, 3, 4)

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

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

Note 3: If the effluent flow past an effluent monitor is known to have stopped 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 RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Gaseous Effluent Monitor Classification Thresholds

Release Point & Monitor	GE	SAE	Alert	NOUE
Vent Stack A VG-RI-179-1 or 2	2.6E+08 μCi/sec	2.6E+07 μCi/sec	2.6E+06 μCi/sec	2.6E+05 μCi/sec
Vent Stack B VG-RI-180-1 or 2	2.0E+08 μCi/sec	2.0E+07 μCi/sec	2.0E+06 μCi/sec	2.0E+05 μCi/sec
Process Vent GW-RI-178-1 or 2	3.5E+08 μCi/sec	3.5E+07 μCi/sec	3.5E+06 μCi/sec	3.5E+05 μCi/sec
Main Steam Line MS-RI-170 (270) MS-RI-171 (271) MS-RI-172 (272)	1.3E+03 mR/hr	1.3E+02 mR/hr	1.3E+01 mR/hr	N/A
TD AFW Pump EXH MS-RI-176 (276)	6.0E+01 mR/hr	6.0E +00 mR/hr	6.0E-01 mR/hr	N/A

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 IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

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

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem adult thyroid CDE was established in consideration of the 1:5 ratio of the 1992 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.

Since dose assessment is based on actual meteorology whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1.

Classification thresholds within Table R-1 were generated using the MIDAS dose assessment code. Inputs to MIDAS use most prevalent meteorological data and expected release point parameters. An assumed one-hour decay since shutdown and a one-hour release duration are applied. Mitigating reduction mechanisms (e.g., decay, sprays, filters) input into MIDAS for each accident type determined the radiological release source term consistent with the guidance provided in NUREG-1228.

The MGPI radiation monitors for 1-GW-RI-178-1 & 2, 1-VG-RI-179-1 & 2 and 1-VG-RI-180-1 & 2 consist of a "normal" (or low) and an "accident" (or high) range device. The "normal" range radiation monitor flowpath is isolated at a predetermined value at which time the "accident" range radiation monitor is automatically aligned for operation. The "normal" range radiation monitor must be manually put back in service when flowpath activity trends down.

The calculated values for the Auxiliary Feedwater Pump Turbine (AFWPT) Exhaust as measured by the MGPI (i.e., Mirion Technologies) radiation monitors were slightly higher for Unit 2 than Unit 1, but within the margin of error for the radiological calculation. The Unit 1 value was used in Table R-1 for both Units 1 and 2 to simplify the table and to eliminate possibility of human error due to reading the wrong unit's value (ref. 1). Therefore, a Unit 2 event would be classified at a slightly lower value than calculated.

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

Reference(s):

1. RP 08-22, "North Anna Abnormal Rad Release Gaseous EAL Thresholds based on NEI 99-01", Rev. 6
2. DC NA-11-01082, "Main Steam Radiation Monitor Replacement"
3. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.2 Alert

Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem adult thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

Basis:

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

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

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem adult thyroid CDE was established in consideration of the 1:5 ratio of the 1992 EPA PAG for TEDE and thyroid CDE.

Since dose assessment is based on actual meteorology whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1. Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible.

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

Reference(s):

1. EPIP-4.01, "Radiological Assessment Director Controlling Procedure"
2. EPIP-4.03, "Dose Assessment Team Controlling Procedure"
3. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.3	**Alert**
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem adult thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure (Notes 1, 2)	

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

Basis:

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

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

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 EPA PAG for TEDE and thyroid CDE.

This EAL is assessed per the ODCM (ref. 1). ODCM software can be used to produce a dose to the maximum individual.

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

Reference(s):

1. VPAP-2103N, "Offsite Dose Calculation Manual (North Anna)"
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.4 Alert
Field survey results indicate **<u>EITHER</u>** 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 adult thyroid CDE > 50 mrem for 60 min. of inhalation.
(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

Basis:

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

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

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 EPA PAG for TEDE and thyroid CDE.

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

Reference(s):

1. EPIP-4.16, "Offsite Monitoring"

2. EPIP-4.01, "Radiological Assessment Director Controlling Procedure"
3. EPIP-4.03, "Dose Assessment Team Controlling Procedure"
4. EPIP-4.34, "Field Team Radio Operator Instructions"
5. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.1 Site Area Emergency

Reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥15 min. (Notes 1, 2, 3, 4)

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

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

Note 3: If the effluent flow past an effluent monitor is known to have stopped 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 RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Gaseous Effluent Monitor Classification Thresholds

Release Point & Monitor	GE	SAE	Alert	NOUE
Vent Stack A VG-RI-179-1 or 2	2.6E+08 µCi/sec	2.6E+07 µCi/sec	2.6E+06 µCi/sec	2.6E+05 µCi/sec
Vent Stack B VG-RI-180-1 or 2	2.0E+08 µCi/sec	2.0E+07 µCi/sec	2.0E+06 µCi/sec	2.0E+05 µCi/sec
Process Vent GW-RI-178-1 or 2	3.5E+08 µCi/sec	3.5E+07 µCi/sec	3.5E+06 µCi/sec	3.5E+05 µCi/sec
Main Steam Line MS-RI-170 (270) MS-RI-171 (271) MS-RI-172 (272)	1.3E+03 mR/hr	1.3E+02 mR/hr	1.3E+01 mR/hr	N/A
TD AFW Pump EXH MS-RI-176 (276)	6.0E+01 mR/hr	6.0E +00 mR/hr	6.0E-01 mR/hr	N/A

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

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

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 EPA PAG for TEDE and thyroid CDE.

Since dose assessment is based on actual meteorology whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1.

Classification thresholds within Table R-1 were generated using the MIDAS dose assessment code. Inputs to MIDAS use most prevalent meteorological data and expected release point parameters. An assumed one-hour decay since shutdown and a one-hour release duration are applied. Mitigating reduction mechanisms (e.g., decay, sprays, filters) input into MIDAS for each accident type determined the radiological release source term consistent with the guidance provided in NUREG-1228.

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.

The MGPI radiation monitors for 1-GW-RI-178-1 & 2, 1-VG-RI-179-1 & 2 and 1-VG-RI-180-1 & 2 consist of a "normal" (or low) and an "accident" (or high) range device. The "normal" range radiation monitor flowpath is isolated at a predetermined value at which time the "accident" range radiation monitor is automatically aligned for operation. The "normal" range radiation monitor must be manually put back in service when flowpath activity trends down.

The calculated values for the Auxiliary Feedwater Pump Turbine (AFWPT) Exhaust as measured by the MGPI (i.e., Mirion Technologies) radiation monitors were slightly higher for Unit 2 than Unit 1, but within the margin of error for the radiological calculation. The Unit 1 value was used in Table R-1 for both Units 1 and 2 to simplify the table and to eliminate possibility of human error due to reading the wrong unit's value (ref. 1). Therefore, a Unit 2 event would be classified at a slightly lower value than calculated.

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

Reference(s):

1. RP 08-22, "North Anna Abnormal Rad Release Gaseous EAL Thresholds based on NEI 99-01", Rev. 6
2. DC NA-11-01082, "Main Steam Radiation Monitor Replacement"
3. NEI 99-01 AS1

Category:	R – Abnormal Rad Levels / Rad Effluent
Subcategory:	1 – Radiological Effluent
Initiating Condition:	Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem adult thyroid CDE

EAL:

RS1.2 Site Area Emergency

Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem adult thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

Basis:

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

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

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 EPA PAG for TEDE and thyroid CDE.

Since dose assessment is based on actual meteorology whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1. Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible.

Escalation of the emergency classification level would be via IC AG1.

Reference(s):

1. EPIP-4.01, "Radiological Assessment Director Controlling Procedure"
2. EPIP-4.03, "Dose Assessment Team Controlling Procedure"
3. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.3 Site Area Emergency

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

Basis:

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

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

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 EPA PAG for TEDE and thyroid CDE.

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

Reference(s):

1. EPIP-4.16, “Offsite Monitoring”
2. EPIP-4.01, “Radiological Assessment Director Controlling Procedure”
3. EPIP-4.03, “Dose Assessment Team Controlling Procedure”

4. EPIP-4.34, "Field Team Radio Operator Instructions"
5. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.1 General Emergency

Reading on **any** Table R-1 effluent radiation monitor > column "GE" for ≥15 min. (Notes 1, 2, 3, 4)

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

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

Note 3: If the effluent flow past an effluent monitor is known to have stopped 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 RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Gaseous Effluent Monitor Classification Thresholds

Release Point & Monitor	GE	SAE	Alert	NOUE
Vent Stack A VG-RI-179-1 or 2	2.6E+08 µCi/sec	2.6E+07 µCi/sec	2.6E+06 µCi/sec	2.6E+05 µCi/sec
Vent Stack B VG-RI-180-1 or 2	2.0E+08 µCi/sec	2.0E+07 µCi/sec	2.0E+06 µCi/sec	2.0E+05 µCi/sec
Process Vent GW-RI-178-1 or 2	3.5E+08 µCi/sec	3.5E+07 µCi/sec	3.5E+06 µCi/sec	3.5E+05 µCi/sec
Main Steam Line MS-RI-170 (270) MS-RI-171 (271) MS-RI-172 (272)	1.3E+03 mR/hr	1.3E+02 mR/hr	1.3E+01 mR/hr	N/A
TD AFW Pump EXH MS-RI-176 (276)	6.0E+01 mR/hr	6.0E +00 mR/hr	6.0E-01 mR/hr	N/A

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

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

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 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.

Since dose assessment is based on actual meteorology whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1.

Classification thresholds within Table R-1 were generated using the MIDAS dose assessment code. Inputs to MIDAS use most prevalent meteorological data and expected release point parameters. An assumed one-hour decay since shutdown and a one-hour release duration are applied. Mitigating reduction mechanisms (e.g., decay, sprays, filters) input into MIDAS for each accident type determined the radiological release source term consistent with the guidance provided in NUREG-1228.

The MGPI radiation monitors for 1-GW-RI-178-1 & 2, 1-VG-RI-179-1 & 2 and 1-VG-RI-180-1 & 2 consist of a "normal" (or low) and an "accident" (or high) range device. The "normal" range radiation monitor flowpath is isolated at a predetermined value at which time the "accident" range radiation monitor is automatically aligned for operation. The "normal" range radiation monitor must be manually put back in service when flowpath activity trends down.

The calculated values for the Auxiliary Feedwater Pump Turbine (AFWPT) Exhaust as measured by the MGPI (i.e., Mirion Technologies) radiation monitors were slightly higher for Unit 2 than Unit 1, but within the margin of error for the radiological calculation. The Unit 1 value was used in Table R-1 for both Units 1 and 2 to simplify the table and to eliminate possibility of human error due to reading the wrong unit's value (ref. 1). Therefore, a Unit 2 event would be classified at a slightly lower value than calculated.

Reference(s):

1. RP 08-22, "North Anna Abnormal Rad Release Gaseous EAL Thresholds based on NEI 99-01", Rev. 6
2. DC NA-11-01082, "Main Steam Radiation Monitor Replacement"
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.2	**General Emergency**
Dose assessment using actual meteorology indicates doses > 1,000 mrem TEDE or 5,000 mrem adult thyroid CDE at or beyond the SITE BOUNDARY (Note 4)	

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

Basis:

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

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

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 EPA PAG for TEDE and thyroid CDE.

Since dose assessment is based on actual meteorology whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1. Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible.

Reference(s):

1. EPIP-4.01, "Radiological Assessment Director Controlling Procedure"

2. EPIP-4.03, “Dose Assessment Team Controlling Procedure”
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.3 General Emergency

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

Basis:

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

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

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the 1992 EPA PAG for TEDE and thyroid CDE.

Reference(s):

1. EPIP-4.16, "Offsite Monitoring"
2. EPIP-4.01, "Radiological Assessment Director Controlling Procedure"
3. EPIP-4.03, "Dose Assessment Team Controlling Procedure"
4. EPIP-4.34, "Field Team Radio Operator Instructions"
5. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: UNPLANNED loss of water level above irradiated fuel

EAL:

RU2.1 NOUE

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

- Spent Fuel Pit Lo Level (1E-C6) alarm
- Report of dropping level in refueling cavity or SFP
- Loss of SFP Cooling suction flow

AND

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

- RM-RMS-152 New Fuel Storage Area
- RM-RMS-153 Fuel Pit Bridge
- RM-RMS-162 (262) Manipulator Crane Area (Refueling Mode)
- RM-RMS-163 (263) Reactor Containment Area

Mode Applicability:

All

Definition(s):

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

REFUELING PATHWAY- Refueling cavity, fuel transfer canal, and spent fuel pit (SFP), but **not** including the reactor vessel, comprise the refueling pathway.

Basis:

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

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

The SFP level is remotely monitored by level switches FC-LS-100 (high) and 101 (low). The level switch initiates high and low level annunciators. The SFP WATER LEVEL LOW alarm (window 1E-C6) actuates if SFP level decreases to the 289 ft 4 in. el. Local level indication is provided by a ruled scale mounted on the east side of the counterfort. Normal level is indicated by the 0 mark on the scale and corresponds to 289 ft 10 in. el. or normal SFP level. Level is normally maintained between the 0 in. mark and the +3 in. mark. The low level alarm corresponds to the -6 in. mark (ref. 1, 2).

The Spent Fuel Pool (SFP) wide-range level indication system is available to monitor water level. Two (2) level instruments are installed in the SFP with indicators, 1-FC-LI-105-1, 2 & 2A provided in the Main Control Room and MCR Computer Rooms. The level instruments will provide level indication over the entire span of the SFP from the top of the fuel racks to 10 inches above the normal operating level (ref. 5).

The specified radiation monitors are those expected to see increase area radiation levels as a result of a loss of REFUELING PATHWAY inventory (ref. 4). Increasing radiation indications on these monitors in the absence of indications of decreasing REFUELING PATHWAY level are not classifiable under this EAL.

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 with Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC RA2.

Reference(s):

1. AR 1-E-C6, "Spent Fuel Pit Lo Level"
2. 0-AP-27, "Malfunction of Spent Fuel Pit Systems"
3. 1(2)-AP-52, "Loss of Refueling Cavity Level During Refueling
4. 0-AP-5.1, "Common Unit Radiation Monitoring System"
5. Design Change NA-13-01043, "BDB Spent Fuel Pool Level Instrumentation Installation – Units 1 & 2"
6. NEI 99-01 AU2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.1	**Alert**
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 NAPS ISFSI, Confinement Boundary is defined as the Sealed Surface Storage Cask (SSSC) or NUHOMS Dry Storage 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- Refueling cavity, fuel transfer canal, and spent fuel pit (SFP), but **not** including the reactor vessel, comprise the refueling pathway.

Basis:

This IC addresses events that have caused IMMINENT or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the 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.

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

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

This EAL escalates from RU2.1 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used (e.g., a boil-off curve). Classification of an event using this EAL should be based on the totality of available indications, reports and observations.

While an area radiation monitor could detect an increase in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.

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

Reference(s):

1. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity

AND EITHER:

- VALID Hi-Hi alarm on **any** of the following radiation monitors:
    - RM-RMS-152 New Fuel Storage Area
    - RM-RMS-153 Fuel Pit Bridge
    - RM-RMS-162 (262) Manipulator Crane Area (Refueling Mode)
    - RM-RMS-163 (263) Reactor Containment Area
    - RM-RMS-159 (259) Containment Particulate
    - RM-RMS-160 (260) Containment Area Gas
- VALID Hi alarm on VG-RI-180-1 Vent Stack B Normal Range

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 NAPS ISFSI, Confinement Boundary is defined as the Sealed Surface Storage Cask (SSSC) or NUHOMS Dry Storage Canister (DSC).

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:

The specified radiation monitors are those expected to see increased area radiation levels as a result of damage to irradiated fuel (ref. 1, 2, 3, 4).

This EAL addresses events that have caused actual damage to an irradiated fuel assembly. 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.

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

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

Reference(s):

1. 1(2)-AP-5, "Unit 1(2) Radiation Monitoring System"
2. 0-AP-5.1, "Common Unit Radiation Monitoring System"
3. 0-AP-5.2, "MGP Radiation Monitoring System"
4. 0-AP-30, "Fuel Failure During Handling"
5. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.3 Alert

Lowering of spent fuel pool level to 10 ft. (Level 2) on 1-FC-LI-105-1, 2 or 2A Spent Fuel Pit Wide Range Level

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 ICs RS1 or RS2.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (1-FC-LI-105-1, 1-FC-LI-105-2 and 1-FC-LI-105-2A) capable of identifying normal level (Level 1 –EL 289 ft. 10 in.), SFP level 10 ft. above the top of the fuel racks (Level 2 –EL 274 ft. 8 in.) and SFP level at 1 foot above the top of the fuel racks (Level 3 –EL 265 ft. 8 in.) (ref. 1, 2, 3).

Level	Plant Elevation	1-FC-LI-105-1, 2 or 2A Reading (ft. above top of spent fuel racks)
1	289 ft. 10 in.	25.2 ft.
2	274 ft. 8 in.	10 ft.
3	265 ft. 8 in.	1 ft.

Reference(s):

1. ETE-CPR-2012-0012, "North Anna Units 1 & 2 – Beyond Design Basis FLEX Strategy Basis Document and Final Integration Plan"

2. DC NA-13-01043, “Beyond Design Basis Spent Fuel Pool Level Instrument Installation – North Anna Units 1 & 2”
3. 0-AP-27, “Malfunction of Spent Fuel Pit Systems”
4. 1(2)-AP-52, “Loss of Refueling Cavity Level During Refueling”
5. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RS2.1	**Site Area Emergency**
Lowering of spent fuel pool level to 1 ft. (Level 3) on 1-FC-LI-105-1, 2 or 2A Spent Fuel Pit Wide Range Level	

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 RG1 or RG2.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (1-FC-LI-105-1, 1-FC-LI-105-2 and 1-FC-LI-105-2A) capable of identifying normal level (Level 1 –EL 289 ft. 10 in.), SFP level 10 ft. above the top of the fuel racks (Level 2 –EL 274 ft. 8 in.) and SFP level at 1 foot above the top of the fuel racks (Level 3 –EL 265 ft. 8 in.) (ref. 1, 2, 3).

Level	Plant Elevation	1-FC-LI-105-1, 2 or 2A Reading (ft. above top of spent fuel racks)
1	289 ft. 10 in.	25.2 ft.
2	274 ft. 8 in.	10 ft.
3	265 ft. 8 in.	1 ft.

Reference(s):

1. ETE-CPR-2012-0012, “North Anna Units 1 & 2 – Beyond Design Basis FLEX Strategy Basis Document and Final Integration Plan”
2. DC NA-13-01043, “Beyond Design Basis Spent Fuel Pool Level Instrument Installation – North Anna Units 1 & 2”
3. 0-AP-27. “Malfunction of Spent Fuel Pit Systems”
4. 1(2)-AP-52, “Loss of Refueling Cavity Level During Refueling”
5. NEI 99-01 AS2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RG2.1	**General Emergency**

Spent fuel pool level **cannot** be restored to at least 1 ft. (Level 3) on 1-FC-LI-105-1, 2 or 2A Spent Fuel Pit Wide Range Level for ≥60 min.
(Note 1)

Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be 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 EAL would likely not be met until well after another General Emergency EAL was met; however, it is included to provide classification diversity.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (1-FC-LI-105-1, 1-FC-LI-105-2 and 1-FC-LI-105-2A) capable of identifying normal level (Level 1 –EL 289 ft. 10 in.), SFP level 10 ft. above the top of the fuel racks (Level 2 –EL 274 ft. 8 in.) and SFP level at 1 foot above the top of the fuel racks (Level 3 –EL 265 ft. 8 in.) (ref. 1, 2, 3).

Level	Plant Elevation	1-FC-LI-105-1, 2 or 2A Reading (ft. above top of spent fuel racks)
1	289 ft. 10 in.	25.2 ft.
2	274 ft. 8 in.	10 ft.
3	265 ft. 8 in.	1 ft.

Reference(s):

1. ETE-CPR-2012-0012, "North Anna Units 1 & 2 – Beyond Design Basis FLEX Strategy Basis Document and Final Integration Plan"
2. DC NA-13-01043, "Beyond Design Basis Spent Fuel Pool Level Instrument Installation – North Anna Units 1 & 2"
3. 0-AP-27, "Malfunction of Spent Fuel Pit Systems"
4. 1(2)-AP-52, "Loss of Refueling Cavity Level During Refueling"
5. NEI 99-01 AG2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

RA3.1 Alert

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

- Control Room
- Central Alarm Station (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:

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

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, RM-RMS-157 (ref. 1). 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 that may be used to assess this EAL threshold. Therefore, this threshold is evaluated using local radiation survey for this area.

Reference(s):

1. 0-AP-5.1, "Common Unit Radiation Monitoring System"
2. NEI 99-01 AA3

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

RA3.2 Alert

An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 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 R-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode**
Aux. Building El 274'	1, 2, 3, 4
Instrument Rack Rooms	4
Cable Vault & Tunnels	

Mode Applicability:

1 - Power Operation, 2 – Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective 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 IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The SEM should consider the cause of the increased radiation levels and determine if another IC may be applicable.

For RA3.2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the increased radiation levels. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., installing temporary shielding, requiring use of non-routine protective equipment, requesting an extension in dose limits beyond normal administrative limits).

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).

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

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

Reference(s):

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

Category C – Cold Shutdown / Refueling System Malfunction

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

Category C EALs are directly associated with cold shutdown or refueling system safety functions. Given the variability of plant configurations (e.g., systems out-of-service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The cold shutdown and refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, CONTAINMENT CLOSURE, and fuel clad integrity for the applicable operating modes (5 - Cold Shutdown, 6 - Refueling, DEF – Defueled).

The events of this category pertain to the following subcategories:

1. <u>RCS Level</u>

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

2. <u>Loss of Emergency AC Power</u>

   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 4160V AC emergency buses.

3. <u>RCS Temperature</u>

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

4. <u>Loss of Vital DC Power</u>

   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. <u>Loss of Communications</u>

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

6. <u>Hazardous Event Affecting Safety Systems</u>

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

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory

EAL:

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

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

Mode Applicability:

5 - Cold Shutdown, 6 - 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:

RCS water level less than a required lower limit is meant to be less than the lower end of the level control band being procedurally maintained for the current condition or evolution.

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

With the plant in Refueling mode, RCS water level is normally maintained at or above the reactor vessel flange (ref. 2).

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

Refueling evolutions that 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 NOUE due to the reduced water inventory that is available to keep the core covered.

This EAL recognizes that the minimum required RCS level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is 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. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
2. 1(2)-OP-4.1, "Controlling Procedure for Refueling"
3. 1(2)-AP-17, "Shutdown LOCA"
4. 1(2)-AP-11, "Loss of RHR"
5. 1(2)-AP-52, "Loss of Refueling Cavity Level During Refueling"
6. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory

EAL:

CU1.2 NOUE

RCS water level **cannot** be monitored

AND EITHER:

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

Table C-1 Sumps/Tanks
• Reactor Containment Sump • Pressurizer Relief Tank (PRT) • Primary Drain Transfer Tank (PDTT) • Component Cooling (CC) Surge Tank • Refueling Water Storage Tank (RWST)

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

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

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

Refueling evolutions that 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 NOUE due to the reduced water inventory that is available to keep the core covered.

This EAL addresses a condition where all means to determine RCS level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels (Table C-1) (ref. 1, 2, 3, 4, 5). Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refueling mode, normal means of RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

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

Reference(s):

1. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
2. 1(2)-OP-4.1, "Controlling Procedure for Refueling"
3. 1(2)-AP-17, "Shutdown LOCA"
4. 1(2)-AP-11, "Loss of RHR"
5. 1(2)-AP-52, "Loss of Refueling Cavity Level During Refueling"
6. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Significant Loss of RCS inventory

EAL:

CA1.1	**Alert**
RCS level < minimum required for continued RHR pump operation	

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

None

Basis:

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

For this EAL, a lowering of RCS water level below the specified value(s) indicates that operator actions have not been successful in restoring and maintaining RCS 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. The classification threshold is based on the lowest RCS level that supports continued decay heat removal pump (RHR) operations per procedure (ref. 1, 2, 3, 4).

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

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

Reference(s):

1. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
2. 1(2)-AP-17, "Shutdown LOCA"
3. 1(2)-AP-11, "Loss of RHR"
4. 1(2)-AP-52, "Loss of Refueling Cavity Level During Refueling"
5. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Significant Loss of RCS inventory

EAL:

CA1.2 Alert

RCS water level **cannot** be monitored for ≥15 min. (Note 1)

AND EITHER

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

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

Table C-1 Sumps/Tanks
• Reactor Containment Sump
• Pressurizer Relief Tank (PRT)
• Primary Drain Transfer Tank (PDTT)
• Component Cooling (CC) Surge Tank
• Refueling Water Storage Tank (RWST)

Mode Applicability:

5 - Cold Shutdown, 6 - 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.

Basis:

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

For this EAL, the inability to monitor RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level (Table C-1) changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS (ref 1, 2, 3, 4, 5).

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refueling mode, normal means of RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

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

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

Reference(s):

1. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
2. 1(2)-OP-4.1, "Controlling Procedure for Refueling"
3. 1(2)-AP-17, "Shutdown LOCA"
4. 1(2)-AP-11, "Loss of RHR"
5. 1(2)-AP-52, "Loss of Refueling Cavity Level During Refueling"
6. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.1 Site Area Emergency

With CONTAINMENT CLOSURE **not** established, **any** confirmed loss of inventory indication, Table C-2, with RVLIS full range < 62%

Table C-2 Inventory Loss Confirmatory Indications
• In service Standpipe and Ultrasonic level bottomed out • Decreasing RVLIS level trend • RHR pump amp fluctuations

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions.

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

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

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS/reactor vessel levels of 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 (ref. 1).

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.

When Reactor Vessel water level decreases to 254.625 ft el., water level is six inches below the elevation of the bottom of the RCS hot leg penetration. When Reactor Vessel water level drops significantly below the elevation of the bottom of the RCS hot leg penetration, all sources of RCS injection have failed or are incapable of making up for the inventory loss. Six inches below the elevation of the bottom of the RCS hot leg penetration can be monitored only by RVLIS full range (62.0%). Level monitoring instruments 1-RC-LI-102 (2-RC-LI-202), 1-RC-LI-103, (2-RC-LI-203) 1-RC-LI-105 (2-RC-LI-205) and RVLIS upper range are offscale low when level is below the elevation of the centerline of the RCS loop hot leg penetration (256.333 ft el.).

Table C-2 provides a list of confirmatory indicators for RCS inventory loss. Due to the variability of accuracy and usability of RVLIS while in Cold Shutdown or Refueling Mode, the use of RVLIS for emergency classification purposes is contingent on one or more of the listed confirmatory indications.

The RVLIS full range threshold has been determined as follows (ref. 2, 3, 4):

Component	Elevation (ft)	Radius (in.)	RVLIS Full Range (%)
RCS hot leg centerline	256.333	14.5	63.0
Bottom of RCS hot leg	255.125	NA	A
6 in. below bottom of hot leg	254.625	NA	B
Top of fuel	252.807	NA	61.0

RVLIS span %/ft = 0.56721

A = 61.0% + (Bottom of RCS hot leg - Top of fuel) x RVLIS span
= 62.3%

B = 61.0% + (6 in. below bottom of hot leg - Top of fuel) x RVLIS span
= 62.0%

EAL RVLIS values have been rounded up to the nearest whole percentage point.

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

Reference(s):

1. OU-AA-200, "Shutdown Risk Management"
2. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
3. 1(2)-OP-4.1, "Controlling Procedure for Refueling"
4. 1(2)-AP-17, "Shutdown LOCA"
5. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.2 Site Area Emergency

With CONTAINMENT CLOSURE established, **any** confirmed loss of inventory indication, Table C-2, with RVLIS full range < 61%

Table C-2 Inventory Loss Confirmatory Indications
• In service Standpipe and Ultrasonic level bottomed out • Decreasing RVLIS level trend • RHR pump amp fluctuations

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions.

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

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

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS/reactor vessel levels of 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 (ref. 1).

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.

This level drop can only be remotely monitored by Reactor Vessel Level Instrumentation System (RVLIS). When Reactor Vessel water level drops below RVLIS full range setpoint of 61% (ref. 2), core uncovery is about to occur.

Table C-2 provides a list of confirmatory indicators for RCS inventory loss. Due to the variability of accuracy and usability of RVLIS while in Cold Shutdown or Refueling Mode, the use of RVLIS for emergency classification purposes is contingent on one or more of the listed confirmatory indications.

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

Reference(s):

1. OU-AA-200, "Shutdown Risk Management"
2. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
3. 1(2)-OP-4.1, "Controlling Procedure for Refueling"
4. 1(2)-AP-17, "Shutdown LOCA"
5. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.3 Site Area Emergency

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

AND

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

- UNPLANNED increase in **any** Table C-1 sump or tank level of sufficient magnitude to indicate core uncovery
- Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery
- **Any** containment area radiation monitor reading > 3 R/hr (Refueling Mode)
- Erratic source range monitor indications

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

Table C-1 Sumps/Tanks
• Reactor Containment Sump
• Pressurizer Relief Tank (PRT)
• Primary Drain Transfer Tank (PDTT)
• Component Cooling (CC) Surge Tank
• Refueling Water Storage Tank (RWST)

Mode Applicability:

5 - Cold Shutdown, 6 - 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.

Basis:

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

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

The inability to monitor RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels (Table C-1). Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS (ref. 1, 2, 3).

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refueling mode, normal means of RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory.

In the Refueling mode, the dose rate above the core will rise as water level in the reactor vessel lowers. The dose rate due to this core shine should result in on-scale indications of > 3 R/hr on containment area radiation monitors (ref. 4).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.

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 ICs CG1 or RG1

Reference(s):

1. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
2. 1(2)-OP-4.1, "Controlling Procedure for Refueling"
3. 1(2)-AP-17, "Shutdown LOCA"
4. RA-0078, "Verification of Radiation Monitor Response to Core Uncovery"
5. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CG1.1 General Emergency

Any confirmed loss of inventory indication, Table C-2, with RVLIS full range < 61% for ≥30 min. (Note 1)

AND

Any Containment Challenge indication, Table C-3

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

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

Table C-2 Inventory Loss Confirmatory Indications
• In service Standpipe and Ultrasonic level bottomed out • Decreasing RVLIS level trend • RHR pump amp fluctuations

Table C-3 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • CTMT hydrogen concentration ≥4% • UNPLANNED increase in CTMT pressure

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions.

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:

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

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

Three conditions are associated with a challenge to containment's capability to serve as an effective barrier to fission product release (Table C-3):

1. With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment (ref. 1). If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.
2. The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit of 4%). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to containment integrity.

   In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. However, 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 burn (4%) (ref. 2). If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.
3. Any UNPLANNED rise in containment pressure in the Cold Shutdown or Refueling mode indicates a potential challenge of CONTAINMENT CLOSURE capability. This is due to the potential use of temporary penetration seals, water seals or other closure mechanisms used to support maintenance that are not suitable to withstand a rise in containment pressure. UNPLANNED containment pressure rise indicates CONTAINMENT CLOSURE cannot be assured and the containment cannot be relied upon as a barrier to fission product release.

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*.

This level drop can only be remotely monitored by Reactor Vessel Level Instrumentation System (RVLIS). When Reactor Vessel water level drops below RVLIS full range setpoint of 61%, core uncovery is about to occur.

Table C-2 provides a list of confirmatory indicators for RCS inventory loss. Due to the variability of accuracy and usability of RVLIS while in Cold Shutdown or Refueling Mode, the use of RVLIS for emergency classification purposes is contingent on one or more of the listed confirmatory indications (ref. 3).

EAL RVLIS values have been rounded up to the nearest whole percentage point.

Reference(s):

1. OU-AA-200, "Shutdown Risk Management"
2. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"
3. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
4. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CG1.2 General Emergency

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

AND

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

- UNPLANNED increase in **any** Table C-1 sump or tank level of sufficient magnitude to indicate core uncovery
- Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery
- **Any** containment area radiation monitor reading > 3 R/hr (Refueling Mode)
- Erratic source range monitor indications

AND

Any Containment Challenge indication, Table C-3

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

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

Table C-1 Sumps/Tanks
• Reactor Containment Sump • Pressurizer Relief Tank (PRT) • Primary Drain Transfer Tank (PDTT) • Component Cooling (CC) Surge Tank • Refueling Water Storage Tank (RWST)

Table C-3 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • CTMT hydrogen concentration ≥4% • UNPLANNED increase in CTMT pressure

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions.

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:

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

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

The inability to monitor RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels (Table C-1). Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS (ref. 2, 3, 4).

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refueling mode, normal means of RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory.

In the Refueling mode, the dose rate above the core will rise as water level in the reactor vessel lowers. The dose rate due to this core shine should result in on-scale indications of > 3 R/hr on containment area radiation monitors (ref. 5).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.

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.

Three conditions are associated with a challenge to containment's capability to serve as an effective barrier to fission product release:

1. With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment (ref.1). If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.
2. The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit of 4%). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to containment integrity.

   In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. However, 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 burn (4%) (ref. 6). If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.
3. Any UNPLANNED rise in containment pressure in the Cold Shutdown or Refueling mode indicates a potential challenge of CONTAINMENT CLOSURE capability. This is due to the potential use of temporary penetration seals, water seals or other closure mechanisms used to support maintenance that are not suitable to withstand a rise in containment pressure. UNPLANNED containment pressure rise indicates

CONTAINMENT CLOSURE cannot be assured and the containment cannot be relied upon as a barrier to fission product release.

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. OU-AA-200, "Shutdown Risk Management"
2. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
3. 1(2)-OP-4.1, "Controlling Procedure for Refueling"
4. 1(2)-AP-17, "Shutdown LOCA"
5. RA-0078, "Verification of Radiation Monitor Response to Core Uncovery"
6. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"
7. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CU2.1 NOUE

AC power capability, Table C-4, to Unit 1(2) 4160V emergency buses H and J 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 SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.

Table C-4 AC Power Sources

Offsite:

Unit 1

- Transfer Bus D
- Transfer Bus F
- Station Bus 1B
- Station Bus 2B

Unit 2

- Transfer Bus E
- Transfer Bus F
- Station Bus 2C
- Station Bus 1A

Onsite:

- 1(2)H EDG
- 1(2)J EDG
- AAC (SBO) Diesel Generator (if already aligned)

Mode Applicability:

5 - Cold Shutdown, 6 - 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:

Table C-4 provides a list of offsite and onsite AC electrical power sources credited for this EAL.

Unit 1(2) 4160V emergency buses H and J are the emergency buses (ref. 5).

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

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

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

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

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

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

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down, the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs). The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). Additional bus ties for Unit 1exist between the 1H emergency bus to 1B station service bus and 1J emergency bus to 2B station service bus which can provide a second independent offsite power sources to each Unit 1 emergency bus. Unit 2 emergency busses can be cross tied between the following: 2C station service bus to 2H and 1A station service bus to 2J, which can provide a second independent offsite power source to each Unit 2 emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions. A bus that is powered from the SBO can be credited as being powered from an independent power source. However, since it takes longer than 15 minutes to align the SBO diesel generator, the generator must be "already aligned" to credit it as an AC power source.

Reference(s):

1. 11715-FE-1A, "Main One Line Diagram (Unit 1)"
2. 12050-FE-1A, "Main One Line Diagram (Unit 2)"
3. 1(2)-ECA-0.0, "Loss of All AC Power"
4. 0-AP-10, "Loss of Electrical Power"
5. UFSAR Section 8.3
6. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CA2.1 Alert

Loss of **all** offsite and **all** onsite AC power to Unit 1(2) 4160V emergency buses H and J for ≥15 min. (Note 1)

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

Mode Applicability:

5 - Cold Shutdown, 6 - 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:

For this EAL credit can be taken for any AC power source that has sufficient capability to operate equipment necessary to maintain a safe shutdown condition, such as FLEX generators, provided it can be aligned within the 15 minute classification criteria.

Unit 1(2) 4160V emergency buses H and J are the emergency buses (ref. 5).

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

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

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

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

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down, the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs). The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). Additional bus ties for Unit 1exist between the 1H emergency bus to 1B station service bus and 1J emergency bus to 2B station service bus which can provide a second independent offsite power sources to each Unit 1 emergency bus. Unit 2 emergency busses can be cross tied between the following: 2C station service bus to 2H and 1A station service bus to 2J, which can provide a second independent offsite power source to each Unit 2 emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions. A bus that is powered from the SBO can be credited as being powered from an independent power source. However, since it takes longer than 15 minutes to align the SBO diesel generator, the generator must be "already aligned" to credit it as an AC power source.

This cold condition EAL is equivalent to the hot condition EAL MS1.1.

Reference(s):

1. 11715-FE-1A, "Main One Line Diagram (Unit 1)"
2. 12050-FE-1A, "Main One Line Diagram (Unit 2)"
3. 1(2)-ECA-0.0, "Loss of All AC Power"
4. 0-AP-10, "Loss of Electrical Power"
5. UFSAR Section 8.3
6. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

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

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions.

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 time to 200°F data when in Mode 6 or the RCS is not intact in Mode 5 (ref. 1). If the RCS is intact, classification should be based on the RCS pressure increase criteria of CA3.1. Guidance for calculating RCS time to 200°F is provided in 1(2)-AP-11 Loss of RHR (ref. 2).

This EAL addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit and represents a potential degradation of the level of safety of the plant (ref. 1). If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the SEM 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 an outage, the level in the reactor vessel will normally be maintained at or above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid increase in reactor coolant temperature depending on the time after shutdown (ref. 2).

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. 1(2)-AP-11, "Loss of RHR"
3. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

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

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

Mode Applicability:

5 - Cold Shutdown, 6- Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions.

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. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the SEM should also refer to EAL CA3.1.

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.

RCS level indications include (ref. 2):

- Standpipe level indication 1(2)-RC-LI-102
- Cold Shutdown Level Indicator 1(2)-RC-LI-103
- Independent RCS Level Indicator 1(2)-RC-LI-105
- RVLIS Upper Range Train
- RVLIS Full Range

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.

Reference(s):

1. Technical Specifications Table 1.1-1
2. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
3. 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 increase in RCS temperature to > 200°F for > Table C-5 duration (Notes 1, 12)

<u>OR</u>

UNPLANNED RCS pressure increase > 10 psi (does not apply to solid plant conditions)

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

Note 12: If an RCS heat removal system is in operation within the applicable Table C-5 heat-up duration and RCS temperature is being reduced, the EAL is **not** applicable.

Table C-5 RCS Heat-up Duration Thresholds

RCS Status	CONTAINMENT CLOSURE Status	Heat-up Duration
Intact **<u>AND</u> not** reduced/decreased inventory		60 min.
Not intact **<u>OR</u>** reduced/decreased inventory	Established	20 min.
	Not established	0 min.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to isolate containment to achieve a functional barrier to fission product release during plant shutdown conditions.

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 time to 200°F data when in Mode 6 or the RCS is not intact in Mode 5 (ref. 1). If the RCS is intact, classification should be based on the RCS pressure increase criteria of CA3.1. Guidance for calculating RCS time to 200°F is provided in 1(2)-AP-11 Loss of RHR (ref. 2).

Decreased Inventory is defined as a condition with fuel in the Reactor Vessel and any RCS Loop Stop Valve closed, or RCS water level less than five percent (5%) in the pressurizer. With the Reactor Vessel Head removed and the Reactor Cavity filled to at least 23 feet above the Reactor Vessel Flange, the RCS is not considered to be in a decreased inventory condition (ref. 3).

Reduced Inventory is defined as a condition with fuel in the Reactor Vessel and water level lower than three feet below the Reactor Vessel flange. This corresponds to a plant elevation of 259.8 ft. If reading RCS Level from the MCR on 1(2)-RC-LI-102, RCS Standpipe, Reduced Inventory corresponds to an indicated level of 42 inches (ref. 3).

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 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, or RCS inventory is reduced (e.g., mid-loop operation in PWRs). 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 or is at reduced inventory, and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because 1) the evaporated reactor coolant may be released directly into the containment atmosphere and subsequently to the environment, and 2) there is reduced reactor coolant inventory above the top of irradiated fuel.

The RCS should be assumed to be intact when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals). With the Pressurizer PORV(s) blocked open, the RCS is considered not intact.

The RCS pressure increase threshold provides a pressure-based indication of RCS heat-up in the absence of RCS temperature monitoring capability. 1(2)-RC-PI-1403B and 1(2)-RC-PI-1402B provide RCS narrow range pressure indication (ref. 4, 5).

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

Reference(s):

1. Technical Specifications Table 1.1-1
2. 1(2)-AP-11, "Loss of RHR"
3. 1(2)-OP-5.4, "Draining the Reactor Coolant System"
4. 1-ICP-RC-P1403 (2-ICP-RC-P2403), "Reactor Coolant System Pressure (Wide and Narrow Range) Protection Channel IV Calibration"

5. 1-ICP-RC-P1402 (2-ICP-RC-P2402), "Reactor Coolant System Pressure (Wide and Narrow Range) Protection Channel I Calibration"
6. 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	**NOUE**
Indicated voltage is < 105 VDC on **required** vital 125 VDC battery buses for ≥15 min. (Note 1)	

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

Mode Applicability:

5 - Cold Shutdown, 6 - 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

There are four independent 125 volt DC systems for each unit.

Each system consists of 125 volt DC distribution panels and its respective battery and a battery charger. The batteries 1(2)-I, 1(2)-II, 1(2)-III, and 1(2)-IV supply power only if the battery chargers fail or if the demand exceeds the capacity of the chargers. The batteries are rated for a minimum of two hours (ref. 1, 2).

A battery terminal voltage of 105 volts DC is the minimum voltage required to ensure proper operation of equipment connected to the DC bus (ref. 4).

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

As used in this EAL, "required" means the vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if 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 NOUE. A loss of vital DC power to Train A would not warrant an emergency classification.

The term "required" is meant to be consistent with the requirements of Technical Specifications for the plant shutdown operating modes (ref. 3).

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

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

This cold condition EAL is equivalent to the hot condition EAL MS2.1.

Reference(s):

1. 1(2)-AP-10, "Loss of Electrical Power"
2. UFSAR Section 8.3.2, "Direct Current Power System"
3. Technical Specifications Section 3.8.5, "DC Sources – Shutdown"
4. 0-OP-6.4, "Operation of the SBO Diesel (SBO Event)"
5. 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 NOUE

Loss of **all** Table C-6 onsite communication methods

OR

Loss of **all** Table C-6 State and local agency communication methods

OR

Loss of **all** Table C-6 NRC communication methods

Table C-6 Communication Methods

System	Onsite	State/ Local	NRC
Radio Communications System	X		
Public Address and Intercom System	X		
Private Branch Telephone Exchange (PBX)	X	X	X
Sound Powered Telephone System	X		
Commercial Telephone System		X	X
Automatic Ring Downs (SONET Ring)		X	
Instaphone Loop		X	
Dedicated NRC Communications			X

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, DEF – Defueled

Definition(s):

None

Basis:

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

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

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

The second EAL condition addresses a total loss of the communications methods used to notify all State and local agencies of an emergency declaration. The State and local agencies referred to here are the Commonwealth of Virginia and affected local communities.

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

This cold condition EAL is equivalent to the hot condition EAL MU7.1.

Reference(s):

1. North Anna Power Station Emergency Plan, Section 7.2, "Communications Systems"
2. UFSAR Section 9.5.2, "Communication Systems"
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 second train of the SAFETY SYSTEM needed for the current operating mode
- Event damage has resulted in VISIBLE DAMAGE to the second 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/SEM

Mode Applicability:

5 - Cold Shutdown, 6 - 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.

Basis:

This IC 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 second 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 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.

An event affecting equipment common to two or more trains of a safety system (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the common equipment) should be classified as an Alert under this EAL, as appropriate to the plant mode. By affecting the functionality of multiple trains of a safety system, the loss of the common equipment effectively meets the two-train impact criteria that underlie the EALs and bases.

An event affecting a single-train safety system (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under this EAL because the two-train impact criteria that underlie the EALs and bases would not be met. If an event affects a single-train safety system, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on SEM judgement.

An event that affects two trains of a safety system (e.g., one train has indications of degraded performance and the other VISIBLE DAMAGE) that also has one or more additional trains should be classified as an Alert under this EAL, as appropriate to the plant mode. This approach maintains consistency with the two-train impact criteria that underlie the EALs and bases and is warranted because the event was severe enough to affect the functionality of two trains of a safety system despite plant design criteria associated with system and system train separation and protection. Such an event may have caused other plant impacts that are not immediately apparent.

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

This cold condition EAL is equivalent to the hot condition EAL MA8.1.

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.

A NOUE 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 NAPS ISFSI is located outside the NAPS PLANT PROTECTED AREA but within the OWNER CONTROLLED AREA. Therefore a hostile security event that leads to a potential loss in the level of safety of the ISFSI is a classifiable event under Security category EAL HA4.1.

Category: ISFSI

Subcategory: Confinement Boundary

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 NOUE

Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask > **any** Table E-1 limit

Table E-1 ISFSI Cask Surface Dose Rate Limits

TN-32	TN-32B HBU	HSM-H
• 116 mrem/hr (neutron + gamma) average on top of the cask • 436 mrem/hr (neutron + gamma) average on the side of the cask	• 192 mrem/hr (neutron + gamma) average on top of the cask • 436 mrem/hr (neutron + gamma) average on the side of the cask	• 1,600 mrem/hr at the front bird screen • 4 mrem/hr at the door centerline • 4 mrem/hr at the end shield wall exterior

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY- The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the NAPS ISFSI, Confinement Boundary is defined as the Sealed Surface Storage Cask (SSSC) or 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 IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage.

The existence of “damage” is determined by radiological survey. The specified EAL threshold values correspond to 2 times the TN-32, TN-32B HBU or Horizontal Storage Module (HSM-H) external cask surface dose rate limits (ref. 1, 2). The technical specification multiple of “2 times”, which is also used in Category R IC RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the “on-contact” dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

NAPS utilizes the Transnuclear TN-32/TN-32B HBU dry storage cask system and the NUHOMS HD System (32PTH DSC/HSM-H) dry cask storage system (ref 1).

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

Reference(s):

1. North Anna Power Station ISFSI NRC Certificate of Compliance 1030 Amendment 1, “Technical Specifications and SER (HSM-H)”
2. Technical Specifications and Bases for North Anna ISFSI (TN-32/TN-32B HBU)
3. NEI 99-01 E-HU1

Category F – Fission Product Barrier Degradation

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

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

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

B. Reactor Coolant System Barrier (RCS): The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

C. Containment Barrier (CTMT): The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the Emergency Classification Level (ECL) from an 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

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 Containment Barrier.
- NOUE ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.

- For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC RG1 has been exceeded.
- The fission product barrier thresholds specified within a scheme reflect plant-specific NAPS 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 SEM would have more assurance that there was no immediate need to escalate to a General Emergency.

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **Any** loss or **any** potential loss of either Fuel Clad or RCS

EAL:

FA1.1 Alert

Any loss or **any** potential loss of **EITHER** Fuel Clad or RCS barrier (Table F-1)

Mode Applicability:

1 - Power Operation, 2 – Startup, 3 – Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

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

At the Alert classification level, Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1

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 Standby, 4 - 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 Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds, bases and references.

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

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

At the Site Area Emergency classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, they 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 Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

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

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

- Loss of Fuel Clad, RCS and Containment Barriers
- Loss of Fuel Clad and RCS Barriers with potential loss of Containment Barrier
- Loss of RCS and Containment Barriers with potential loss of Fuel Clad Barrier
- Loss of Fuel Clad and Containment Barriers with potential loss of RCS Barrier

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 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. RCS or SG Tube Leakage

B. Inadequate Heat removal

C. CTMT Radiation / RCS Activity

D. CTMT Integrity or Bypass

E. SEM 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.

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

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

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

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

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad Barrier (FC)		Reactor Coolant System Barrier (RCS)		Containment Barrier (CTMT)	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A RCS or SG Tube Leakage	None	None	1. An automatic or manual Safety Injection (SI) actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	1. UNISOLABLE RCS or SG tube leakage > 150 gpm 2. Integrity-RED Path conditions met	1. A leaking or RUPTURED SG is FAULTED outside of CTMT	None
B Inadequate Heat Removal	1. Core Cooling-RED Path conditions met	1. Core Cooling-ORANGE Path conditions met 2. Heat Sink-RED Path conditions met **AND** Heat sink is required	None	3. Heat Sink-RED Path conditions met **AND** Heat sink is required	None	1. Core Cooling-RED PATH conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C CTMT Radiation / RCS Activity	2. CTMT High Range Radiation Monitor RM-RMS-165/166(265/266) reading > Table F-2 column Fuel Clad Loss 3. Coolant activity > 300 µCi/gm DEI-131 4. Dose rate at 1 ft. from an unpressurized RCS sample ≥Table F-3 5. Sample line dose rate threshold ≥Table F-4 6. With letdown in service, Reactor Coolant Letdown Radiation Monitor CH-RI-128(228) > 7.5E+04 mR/hr	None	2. CTMT High Range Radiation Monitor RM-RMS-165/166(265/266) reading > Table F-2 column RCS Loss	None	None	2. CTMT High Range Radiation Monitor RM-RMS-165/166(265/266) reading > Table F-2 column CTMT Potential Loss
D CTMT Integrity or Bypass	None	None	None	None	2. CTMT isolation (Phase A or B) is required **AND EITHER:** • CTMT integrity has been lost based on SEM judgment • UNISOLABLE pathway from CTMT atmosphere to the environment exists 3. Indications of UNISOLABLE RCS leakage outside of CTMT	3. Containment-RED Path conditions met 4. CTMT hydrogen concentration ≥4% 5. CTMT pressure > 28 psia with < one full train of CTMT heat removal systems (Note 11) operating per design for ≥15 min. (Note 1)
E SEM Judgment	7. **Any** condition in the opinion of the SEM that indicates loss of the fuel clad barrier	3. **Any** condition in the opinion of the SEM that indicates potential loss of the fuel clad barrier	3. **Any** condition in the opinion of the SEM that indicates loss of the RCS barrier	4. **Any** condition in the opinion of the SEM that indicates potential loss of the RCS barrier	4. **Any** condition in the opinion of the SEM that indicates loss of the CTMT barrier	6. **Any** condition in the opinion of the SEM that indicates potential loss of the CTMT barrier

Barrier: Fuel Clad

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

1. Core Cooling-RED Path conditions met

Definition(s):

None

Basis:

This condition indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

The loss threshold is based on meeting either CSFST Core Cooling Red path criteria (ref. 1, 2):

- Core Exit Thermocouple readings ≥1,200 °F.
- Core exit TCs are ≥700°F with RCS subcooling based on core exit TCs ≤25°F [75°F], no RCPs are running, and RVLIS full range is ≤48%

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 2 Core Cooling"
2. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. Core Cooling-ORANGE Path conditions met

Definition(s):

None

Basis:

This condition indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

The potential loss threshold is based on meeting the CSFST Core Cooling Orange Path criteria.

CSFST Core Cooling-ORANGE path is entered if core exit thermocouples (TCs) are < 1,200°F, RCS subcooling based on core exit TCs is ≤25°F [75°F], and either of the following (ref. 1, 2):

- No RCPs are running and either: core exit TCs are ≥700°F and RVLIS full range is > 48%, or core exit TCs are < 700°F and RVLIS full range is ≤48%.
- At least one RCP is running and Reactor Vessel water level is ≤the specified RVLIS dynamic head threshold readings based on the number of RCPs running.

Reactor Vessel Water Level Thresholds		
RVLIS	**No. RCPs**	**Threshold**
Full Range	None	48%
Dynamic Range	3	65%
	2	41%
	1	30%

Consistent with Section 3.2.6 Classification of Transient Conditions, expected short term CSFST Core Cooling-ORANGE path conditions existing prior to successful automatic ECCS actuation following a large break LOCA would not meet the intent of this threshold.

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 2 Core Cooling"
2. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"

3. NEI 99-01 Inadequate Heat Removal Fuel Clad Potential Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. Heat Sink-RED Path conditions met **AND** Heat sink is required

Definition(s):

None

Basis:

The potential loss threshold is based on meeting the CSFST Heat Sink Red Path criteria of both of the following conditions existing (ref. 1):

- Narrow Range levels in all SGs < 11% [22%]
- Total feedwater flow to SGs ≤340 gpm

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

The phrase "and heat sink required" precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. For example, FR-H.1 is entered from CSFST Heat Sink-Red. Step 1 tells the operator to determine if secondary heat sink is required by checking that RCS pressure is greater than any non-faulted SG pressure and RCS T_{hot} is greater than 350°F. If these conditions exist, Heat Sink is required. Otherwise, the operator is to either go to the procedure and step in effect or place RHR in service for heat removal. For large LOCA events inside the Containment, the SGs are irrelevant because heat removal through the containment heat removal systems takes place. Therefore, Heat Sink Red should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification. (ref. 1, 2).

Meeting this threshold results in a Site Area Emergency because this threshold is identical to RCS Barrier Potential Loss threshold B.3; both will be met. This condition warrants a Site Area Emergency declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase RCS pressure to the point where mass will be lost from the system.

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 3 Heat Sink"
2. 1(2)-FR-H.1, "Response to Loss of Secondary Heat Sink"
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Potential Loss 2.B

Barrier: Fuel Clad

Category: C. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. CTMT high range radiation monitor RM-RMS-165/166(265/266) reading > Table F-2 column Fuel Clad Loss

Table F-2 CTMT High Range Radiation Monitor Barrier Thresholds RM-RMS-165/166(265/266)			
Time > Shutdown (hrs)	**Fuel Clad Loss (R/hr)**	**RCS Loss (R/hr)**	**CTMT Potential Loss (R/hr)**
≤2	125	5	500
> 2 – ≤4	85	5	340
> 4 – ≤6	45	5	180
> 8 – ≤14	20	5	80
> 14	10	5	40

Definition(s):

None

Basis:

Containment radiation monitor readings greater than the Table F-2 Fuel Clad Loss column threshold indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading is derived assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 5% clad failure into the containment atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within Technical Specifications and are therefore indicative of fuel damage (approximately 5 % clad failure depending on core inventory and RCS volume) (ref. 1, 2).

Time after shutdown values are provided to account for radioactive decay.

The values specified in Table F-2 were developed using a method to minimize error (+/-) for the threshold value within each defined time period. Time periods were chosen to fit monitor response (fast changes in response early following reactor shutdown are broken up into smaller time periods to better approximate expected change). Values were chosen within each time period to minimize error (<50%) to the highest and lowest response within the range.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold C.2 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 RCS Activity / Containment Radiation.

Reference(s):

1. Calculation RA-0064, "Expected Containment High Range Radiation Monitor Response to a LOCA Based on Fuel Rod Gap Fractions Defined in NUREG 1228"
2. NEI 99-01 CTMT Radiation / RCS Activity FC Loss 3.A

Barrier: Fuel Clad

Category: C. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

3. Coolant activity > 300 µCi/gm DEI-131

Definition(s):

None

Basis:

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

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 RCS Activity / Containment Radiation.

Reference(s):

1. NEI 99-01 CTMT Radiation / RCS Activity FC Loss 3.B

Barrier: Fuel Clad

Category: C. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

4. Dose rate at 1 ft. from an unpressurized RCS sample ≥Table F-3

Table F-3 FC Loss Coolant Activity Dose Rates	
Time > Shutdown (hrs)	**mR/hr/ml**
≤2	15
> 2 – ≤8	8
> 8	3

Definition(s):

None

Basis:

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

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. This EAL provides the ability to take a dose rate off of an RCS sample to determine fuel clad barrier loss, without the need to analyze the sample before making this determination. This EAL saves significant time by allowing evaluation of contained radioactivity within the RCS by a direct dose rate measurement.

Per Engineering Calculation RA-0059, dose rate is assumed to result from radioactive iodines (I-131 thru I-135) in RCS in concentrations corresponding to the loss of 5% of gap radioactivity of the core. For 5% loss of gap radioactivity (~300 μCi/gm DEI-131), 2% of the core inventory of radioactive iodines are assumed to be contained in the gap. The values contained in Table F-3 (FC Loss Coolant Activity Dose Rates) represent expected one foot dose rates per ml of sample based on time since reactor shutdown to the time when the sample is taken. The expected dose rate is a near linear relationship with the volume of the sample, so any volume collected can be determined by dividing the measured dose rate by the sample volume and comparing to the threshold value from Table F-3 for the applicable time frame. These dose rates assume no ECCS injection so there is no dilution credited which would vary coolant volume. Values in the table have been rounded for ease of use. The > 8 hour threshold is conservative up to 24 hours following reactor shutdown. After 24 hours, the expected response from radioactive iodine levels off. Therefore, the value shown for > 8 hours applies for all samples taken 8 hours or more since reactor shutdown (ref. 1, 2).

The values specified in Table F-3 were developed using a method to minimize error (+/-) for the threshold value within each defined time period. Values were chosen to minimize error from the highest to lowest dose rate within each range.

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

Reference(s):

1. Calculation RA-0059, "Detector Response to an RCS Sample for EAL Classification of Fuel Clad Degradation and Barrier Loss"
2. NEI 99-01 CTMT Radiation / RCS Activity FC Loss 3.B

Barrier: Fuel Clad

Category: C. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

5. Sample line dose rate threshold ≥Table F-4

Table F-4 FC Loss RCS Sample Line Dose Rates	
Time > Shutdown (hrs)	**R/hr**
≤2	4
> 2 – ≤8	2
> 8	1

Definition(s):

None

Basis:

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

Per Engineering Calculation RA-0079, dose rate is assumed to result from radioactive iodines in the RCS in concentrations corresponding to the loss of 5% of gap radioactivity of the core. The values contained in Table F-4 (FC Loss RCS Sample Line Dose Rates) represent fuel clad failure thresholds when measured approximately 2" from the outside of the RCS hot leg sample line. RCS sample line locations have been predetermined for use with this EAL. Other RCS lines could be used if analyzed on a case-by-case basis. Values in the table have been rounded for ease of use. The sample line dose rates have been calculated for various time ranges after shutdown (ref. 1).

The values specified in Table F-4 were developed using a method to minimize error (+/-) for the threshold value within each defined time period. Values were chosen to minimize error from the highest to lowest dose rate within each range.

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

Reference(s):

1. Engineering Calculation RA-0079
2. NEI 99-01 CTMT Radiation / RCS Activity FC Loss 3.B

Barrier: Fuel Clad

Category: C. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

6. With letdown in service, Reactor Coolant Letdown Radiation Monitor 1(2)-CH-RI-128(228) > 7.5E+04 mrem/hr

Definition(s):

None

Basis:

This threshold indicates that RCS radioactivity concentration is greater than 300 μCi/gm DEI-131 (ref. 1). Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

A portion of the letdown stream flows past radiation monitors 1(2)-CH-RM-128(228) to detect fission product activity in the reactor coolant and warn of a potential fuel element failure (ref. 2).

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

Reference(s):

1. Calculation No. PA-0234, Rev. 1 "Post Accident Letdown Radiation Monitor Response for North Anna"
2. UFSAR Section 11.4.2.15, "Reactor Coolant Letdown Gross Activity Monitors"
3. NEI 99-01 CTMT Radiation / RCS Activity FC Loss 3.B

Barrier: Fuel Clad

Category: C. CTMT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. SEM Judgment

Degradation Threat: Loss

Threshold:

7. **Any** condition in the opinion of the SEM 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 SEM 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. SEM Judgment

Degradation Threat: Potential Loss

Threshold:

3. **Any** condition in the opinion of the SEM 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 SEM in determining whether the Fuel Clad barrier is potentially lost. The SEM 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. RCS or S/G Tube Leakage

Degradation Threat: Loss

Threshold:

1. An automatic or manual Safety Injection (SI) actuation required by **EITHER**:
   - UNISOLABLE RCS leakage
   - SG tube RUPTURE

Definition(s):

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

RUPTURE - The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

Basis:

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

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

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

This threshold does not apply to a Safety Injection (SI) actuation not caused by excessive RCS leakage (i.e., steamline ΔP or high steam flow) (ref. 1).

If EOPs direct operators to open the Pressurizer pressure relief valves to implement a core cooling strategy (i.e., a "feed and bleed" cooldown), then there will exist a reactor coolant flow path from the RCS, past the "pressurizer safety and relief valves" and into the containment that operators cannot isolate without compromising the effectiveness of the strategy (i.e., for the strategy to be effective, the valves must be kept in the open position); therefore, the flow through the pressure relief line is UNISOLABLE. In this case, the ability of the RCS pressure boundary to serve as an effective barrier to a release of fission products has been eliminated and thus this condition constitutes a loss of the RCS barrier.

Reference(s):

1. 1(2)-E-0, "Reactor Trip or Safety Injection"
2. 1(2)-E-3, "Steam Generator Tube Rupture"

3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Loss 1.A

Barrier: Reactor Coolant System

Category: A. RCS or S/G Tube Leakage

Degradation Threat: Potential Loss

Threshold:

1. UNISOLABLE RCS or SG tube leakage > 150 gpm

Definition(s):

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

Basis:

This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging pump, but an SI actuation has not occurred. The threshold is met when RCS leakage is determined to exceed 150 gpm excluding normal reductions in RCS inventory such as letdown and RCP seal leakoff (ref.1).

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

If the leaking steam generator (> 150 gpm) is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold A.1 will also be met.

Reference(s):

1. NAPS FSAR Table 9.3-5, "Principal Component Data Summary"
2. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.A

Barrier: Reactor Coolant System

Category: A. RCS or S/G Tube Leakage

Degradation Threat: Potential Loss

Threshold:

2. Integrity-RED Path conditions met

Definition(s):

None

Basis:

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

The potential loss threshold is defined by the CSFST Integrity - RED path. CSFST Integrity - Red Path plant conditions (> 100°F/hr cold leg cooldown) and associated PTS Limit A Curve indicates an extreme challenge to the safety function when plant parameters are to the left of the limit curve following excessive RCS cooldown under pressure (ref. 1).

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees Attachment 4 Integrity"
2. 1(2)-FR-P.1, "Response to Imminent Pressurized Thermal Shock Condition"
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.B

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. Heat Sink-RED Path conditions met

   AND

   Heat sink is required

Definition(s):

None

Basis:

The potential loss threshold is based on meeting the CSFST Heat Sink Red Path criteria of both of the following conditions existing (ref. 1):

- Narrow Range levels in all SGs < 11% [22%]
- Total feedwater flow to SGs ≤340 gpm

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

The phrase "and heat sink required" precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. For example, FR-H.1 is entered from CSFST Heat Sink-Red. Step 1 tells the operator to determine if heat sink is required by checking that RCS pressure is greater than any non-faulted SG pressure and RCS T_{hot} is greater than 350°F. If these conditions exist, Heat Sink is required. Otherwise, the operator is to either go to the procedure and step in effect or place RHR in service for heat removal. For large LOCA events inside the Containment, the SGs are irrelevant because heat removal through the containment heat removal systems takes place. Therefore, Heat Sink Red is not applicable and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification. (ref. 1, 2).

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

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees Attachment 3 Heat Sink"
2. 1(2)-FR-H.1, "Response to Loss of Secondary Heat Sink"
3. NEI 99-01 Inadequate Heat Removal RCS Loss 2.B

Barrier: Reactor Coolant System

Category: C. CTMT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

2. CTMT high range radiation monitor RM-RMS-165/166(265/266) reading > Table F-2 column RCS Loss

Table F-2 CTMT High Range Radiation Monitor Barrier Thresholds RM-RMS-165/166(265/266)			
Time > Shutdown (hrs)	**Fuel Clad Loss (R/hr)**	**RCS Loss (R/hr)**	**CTMT Potential Loss (R/hr)**
≤2	125	5	500
> 2 – ≤4	85	5	340
> 4 – ≤6	45	5	180
> 8 – ≤14	20	5	80
> 14	10	5	40

Definition(s):

None

Basis:

A reading > 5 R/hr (minimum practical reading) on RM-RMS-165/166(265/266) is indicative of a breach in the RCS barrier (ref. 1, 2).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad barrier loss threshold C.2 since it indicates a loss of the RCS Barrier only.

Because of the very high fuel clad integrity, only small amounts of noble gases would be dissolved in the primary coolant. Conservative estimates indicated that the readings from release of the normal RCS inventory would be below normal readings on the monitor while the station was operating. Therefore, a value 5 times the normal containment radiation monitor RM-RMS-165/166(265/266) reading of ~ 1 R/hr is used. The reading is less than that specified for fuel cladding barrier loss because no damage to the fuel cladding is assumed. Only leakage from the RCS is assumed for this barrier loss threshold. The value is high enough to preclude erroneous classification of barrier loss due to normal plant operations and is the lowest readable value on the monitors (ref. 1).

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

Reference(s):

1. Calculation RA-0064, "Expected Containment High Range Radiation Monitor Response to a LOCA Based on Fuel Rod Gap Fractions Defined in NUREG 1228"
2. NEI 99-01 CMT Radiation / RCS Activity RCS Loss 3.A

Barrier: Reactor Coolant System

Category: C. CTMT Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. SEM Judgment

Degradation Threat: Loss

Threshold:

3. **Any** condition in the opinion of the SEM that indicates loss of the RCS barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that may be used by the SEM 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: E. SEM Judgment

Degradation Threat: Potential Loss

Threshold:

4. **Any** condition in the opinion of the SEM that indicates potential loss of the RCS barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that may be used by the SEM in determining whether the RCS barrier is potentially lost. The SEM 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: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. A leaking or RUPTURED SG is FAULTED outside of CTMT

Definition(s):

FAULTED - The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.

RUPTURED - The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

Basis:

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

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

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC MU4 for the fuel clad barrier (i.e., RCS activity values) and IC MU5 for the RCS barrier (i.e., RCS leak rate values).

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

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

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

The emergency classification levels resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

	Affected SG is FAULTED Outside of Containment?	
P-to-S Leak Rate	**Yes**	**No**
Less than or equal to 25 gpm	No classification	No classification
Greater than 25 gpm	NOUE per MU5.1	NOUE per MU5.1
> 150 gpm (*RCS Barrier Potential Loss*)	Site Area Emergency per FS1.1	Alert per FA1.1
Requires an automatic or manual ECCS (SI) actuation (*RCS Barrier Loss*)	Site Area Emergency per FS1.1	Alert per FA1.1

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

Reference(s):

1. 1-E-2 (2-E-2), “Faulted Steam Generator Isolation”
2. 1-E-3 (2-E-3), “Steam Generator Tube Rupture”
3. NEI 99-01 RCS or SG Tube Leakage Containment Loss 1.A

Barrier: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

None

Barrier: Containment

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. Core Cooling-RED Path conditions met

   AND

   Restoration procedures **not** effective within 15 min. (Note 1)

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

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:

The potential loss threshold is based on meeting either CSFST Core Cooling Red Path criteria (ref. 1, 2):

- Core Exit Thermocouple readings ≥1,200 °F.
- Core exit TCs are ≥700°F with RCS subcooling based on core exit TCs ≤25°F [75°F], no RCPs are running, and RVLIS full range is ≤48%

and restoration procedures not effective within 15 minutes.

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

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

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

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 2 Core Cooling"
2. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"
3. NEI 99-01 Inadequate Heat Removal Containment Potential Loss 2.A

Barrier: Containment

Category: C. CTMT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: C. CTMT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

2. CTMT high range radiation monitor RM-RMS-165/166(265/266) reading > Table F-2 column CTMT Potential Loss

Table F-2 CTMT High Range Radiation Monitor Barrier Thresholds RM-RMS-165/166(265/266)

Time > Shutdown (hrs)	Fuel Clad Loss (R/hr)	RCS Loss (R/hr)	CTMT Potential Loss (R/hr)
≤2	125	5	500
> 2 – ≤4	85	5	340
> 4 – ≤6	45	5	180
> 8 – ≤14	20	5	80
> 14	10	5	40

Definition(s):

None

Basis:

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds (ref. 1).

Time after shutdown values are provided to account for radioactive decay.

The values specified in Table F-2 were developed using a method to minimize error (+/-) for the threshold value within each defined time period. Time periods were chosen to fit monitor response (fast changes in response early following reactor shutdown are broken up into smaller time periods to better approximate expected change). Values were chosen within each time period to minimize error (<50%) to the highest and lowest response within the range.

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.

Reference(s):

1. Calculation RA-0064, "Expected Containment High Range Radiation Monitor Response to a LOCA Based on Fuel Rod Gap Fractions Defined in NUREG 1228"
2. NEI 99-01 CMT Radiation / RCS Activity Containment Potential Loss 3.A

Barrier: Containment

Category: D. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

2. CTMT isolation (Phase A or B) is required

   AND EITHER:

   - CTMT integrity has been lost based on SEM judgment
   - UNISOLABLE pathway from CTMT atmosphere to the environment exists

Definition(s):

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

Basis:

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold A.1. Therefore this threshold is not applicable to steam generator tube leakage.

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

First Threshold – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the SEM will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure 1. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

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

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

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

Refer to the top piping run of Figure 1. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

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

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

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

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

Reference(s):

1. UFSAR Section 6.2.4, "Containment Isolation System"
2. NEI 99-01 CMT Integrity or Bypass Containment Loss 4.A

Barrier: Containment

Category: D. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

3. Indications of UNISOLABLE RCS leakage outside of CTMT

Definition(s):

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

Basis:

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

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Containment Loss Threshold A.1. Therefore this threshold is not applicable to steam generator tube leakage.

This threshold **does not** apply to an UNISOLABLE RSHX tube leak outside containment. Such leaks are properly addressed under the category R radiological release based EALs.

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

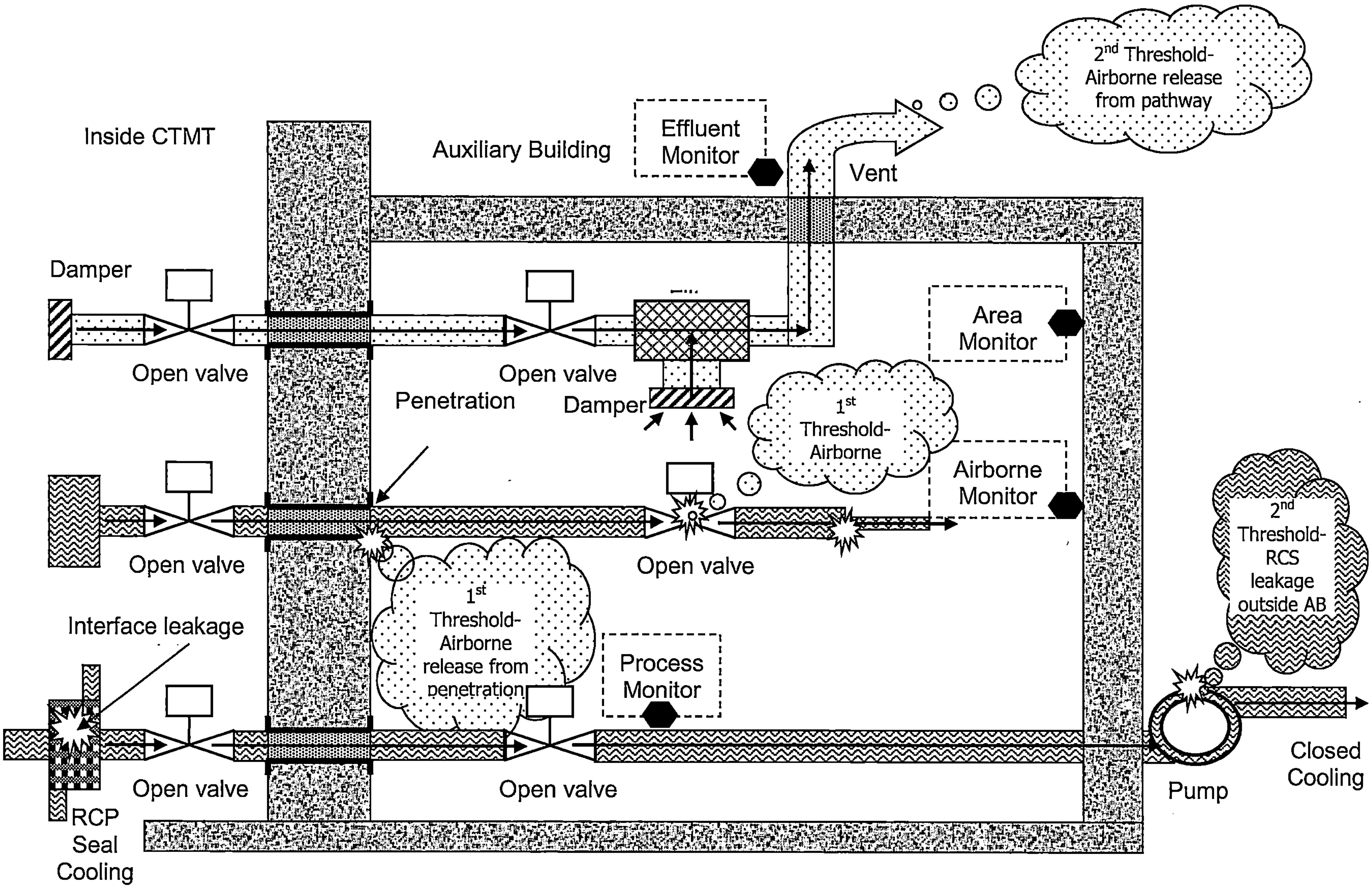
Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of RCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside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

Refer to the middle piping run of Figure 1. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause loss threshold D.2 to be met as well.

Reference(s):

1. NEI 99-01 CMT Integrity or Bypass Containment Loss 4.B

Figure 1: Containment Integrity or Bypass Examples



Barrier: Containment

Category: D. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

3. Containment RED Path conditions met.

Definition(s):

None

Basis:

CSFST Containment RED Path conditions are met if containment pressure exceeds its design pressure. If containment pressure exceeds the design pressure of 60 psia (ref. 1, 2), there exists a potential to lose the containment barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the RCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a Site Area Emergency and General Emergency since there is now a potential to lose the third barrier.

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 5 Containment"
2. UFSAR Section 6.2
3. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.A

Barrier: Containment

Category: D. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

4. CTMT hydrogen concentration ≥4%

Definition(s):

None

Basis:

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

A containment hydrogen concentration of 4% conservatively represents the lowest threshold for flammability in the presence of oxygen (ref. 1,2).

Containment hydrogen analyzers 1-HC-H2A-101 and 2-HC-H2A-201 display hydrogen concentration on PAMC-1 and PAMC-2 with a range of 0 - 10% (ref. 3).

Reference(s):

1. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"
2. SAMG CA-3, "Calculation Aid Number 3 - Hydrogen Flammability in Containment:\"
3. UFSAR Table 7.5-2
4. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.B

Barrier: Containment

Category: D. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

5. CTMT pressure > 28 psia with < one full train of CTMT depressurization equipment (Note 11) operating per design for ≥15 min. (Note 1)

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

Note 11: One full train of containment depressurization equipment consist of one Quench Spray (QS) System and one Recirculation Spray (RS) System from either train operating together.

Definition(s):

None

Basis:

This threshold describes a condition where containment pressure is greater than the setpoint (28 psia) (ref. 3, 4) at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design (ref. 1, 2). The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems (e.g., containment sprays but not including containment venting strategies) are either lost or performing in a degraded manner.

The Quench Spray (QS) System, operating in conjunction with the Recirculation Spray (RS) System, is designed to cool and depressurize the containment structure to less than 2.0 psig in one hour and sub-atmospheric pressure in less than 6 hours following a Design Basis Accident. The combination of required equipment can be obtained from using equipment on either emergency busses in order to meet the “one full train” requirement (ref. 1, 2).

Reference(s):

1. Technical Specifications Section B 3.6.6, “Quench Spray (QS) System”
2. Technical Specifications Section B 3.6.,7 “Recirculation Spray (RS) System”
3. 1(2)-F-0, “Critical Safety Function Status Trees, Attachment 5 Containment”
4. 1(2)-FR-Z.1, “Response to High Containment Pressure”
5. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.C

Barrier: Containment

Category: E. SEM Judgment

Degradation Threat: Loss

Threshold:

4. **Any** condition in the opinion of the SEM that indicates loss of the CTMT barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that may be used by the SEM in determining whether the containment barrier is lost.

Reference(s):

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

Barrier: Containment

Category: E. SEM Judgment

Degradation Threat: Potential Loss

Threshold:

6. **Any** condition in the opinion of the SEM that indicates potential loss of the CTMT barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that may be used by the SEM in determining whether the containment barrier is potentially lost. The SEM 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 Containment 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 PLANT 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.

7. SEM 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 SEM the latitude to classify emergency conditions consistent with the established classification criteria based upon SEM judgment.

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1 NOUE

A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by NAPS 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 NAPS 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 NAPS. 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) - The entire area contiguous to the PLANT PROTECTED AREA, owned by the Company and designated to be controlled for security reasons.

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

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

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.

Basis:

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR 73.71 or 10 CFR 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1 and HS1. Guidance on assessing Security Conditions is included in the Security Contingency Implementing Procedures (SCIP). The SCIPs are implementing procedures for the Station Safeguards Contingency Plan.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1, 2, 3). Classification of these events will initiate appropriate threat-related notifications to plant personnel and State and local agencies.

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

The second threshold addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the Millstone, North Anna and Surry Power Stations' Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program (ref. 1) and associated Security Plan Implementing Procedures (SCIP).

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 0-AP-9 Station Security 9 – Operations Response or 0-AP-9.01 Station Security Air Threat – Operations Response (ref. 2, 3).

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 NAPS (ref. 1).

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

Reference(s):

1. Millstone, North Anna and Surry Power Stations' Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program
2. 0-AP-9, "Station Security – Operations Response"
3. 0-AP-9.01, "Station Security Air Threat – Operations Response"
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HA1.1 Alert

A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by NAPS 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 NAPS 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 NAPS. 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 - The entire area contiguous to the PLANT PROTECTED AREA, owned by the Company and designated to be controlled for security reasons.

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

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PLANT 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 (ref. 1, 2, 3).

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 State and local agencies, 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 10 CFR 73.71 or 10 CFR 50.72.

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

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 State and local agencies are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with 0-AP-9 Station Security – Operations Response or 0-AP-9.01 Station Security Air Threat – Operations Response (ref. 2, 3).

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

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

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

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

Reference(s):

1. Millstone, North Anna and Surry Power Stations' Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program

2. 0-AP-9, "Station Security – Operations Response"
3. 0-AP-9.01, "Station Security Air Threat – Operations Response"
4. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the PLANT PROTECTED AREA

EAL:

HS1.1	**Site Area Emergency**
A HOSTILE ACTION is occurring or has occurred within the PLANT PROTECTED AREA as reported by NAPS 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 NAPS 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 NAPS. 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 - The entire area contiguous to the PLANT PROTECTED AREA, owned by the Company and designated to be controlled for security reasons.

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

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

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the PLANT 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, 2, 3).

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

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

Reference(s):

1. Millstone, North Anna and Surry Power Stations' Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program
2. 0-AP-9, "Station Security – Operations Response"
3. 0-AP-9.01, "Station Security Air Threat – Operations Response"
4. 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	**NOUE**
Seismic event > OBE (0.06g horizontal or 0.04g vertical) as indicated by "OBE EXCEEDED" indicator illuminated on the SYSCOM Network Control Center (NCC)	

Mode Applicability:

All

Definition(s):

None

Basis:

0-AP-36 Seismic Event provides the guidance for determining if the OBE earthquake threshold is exceeded (horizontal or vertical) and any required response actions. (ref. 2).

Ground motion acceleration of 0.06g horizontal or 0.04g vertical is the Operating Basis Earthquake for NAPS (ref. 1).

Ground motion acceleration at the OBE is unmistakably a "felt" earthquake and is significantly greater than the ground motion acceleration required to activate the Event Indicator on the Strong Motion Accelerograph (SMA) which, in turn, activates annunciator 1A-B4, Earthquake System Trigger, in the Control Room. The "OBE EXCEEDED" indicator illuminates on the SYSCOM Network Control Center (NCC) if site OBE ground acceleration is exceeded (ref. 2).

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 significant seismic event (e.g., lateral accelerations in excess of 0.06g). The Shift Manager 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.

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE). An earthquake greater than an OBE but less than a Design Basis Earthquake (DBE) 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.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or MA9.

Reference(s):

1. UFSAR Section 2.5.2.6
2. 0-AP-36, "Seismic Event"
3. 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	**NOUE**
A tornado strike within the PLANT PROTECTED AREA	

Mode Applicability:

All

Definition(s):

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

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 PLANT PROTECTED AREA.

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

If damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under IC CA6 or MA9.

A tornado striking (touching down) within the PLANT PROTECTED AREA warrants declaration of an NOUE 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. 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	**NOUE**
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.

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 (ref. 1, 2).

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

Refer to EAL CA6.1 or MA9.1 for internal flooding affecting more than one SAFETY SYSTEM train.

Reference(s):

1. 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 NOUE

Movement of personnel within the PLANT PROTECTED AREA is IMPEDED due to an event external to the PLANT 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).

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

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 PLANT PROTECTED AREA and of sufficient magnitude to IMPEDE the movement of personnel within the PLANT PROTECTED AREA.

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

Reference(s):

1. 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 NOUE

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

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

Mode Applicability:

All

Definition(s):

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 on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site FLOODING caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.

This EAL is not intended 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 R, F, M or C.

Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.1 NOUE

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

Table H-1 NAPS Fire Areas
• Cable Vaults & Tunnels
• Emergency Switchgear Rooms
• Emergency Diesel Generator Rooms
• Reactor Containment
• Quench Spray Pump Houses
• Safeguards Area
• Main Steam Valve House
• Cable Spreading Rooms
• Control Room
• CR Chiller Rooms
• Auxiliary / Fuel / Decontamination Buildings
• Fuel Oil Pump House Room A or B
• Service Water Pump House and Valve House
• Intake Structure Control House
• Auxiliary Service Water Pump House
• Auxiliary Feedwater Pump House
• Turbine Building

Mode Applicability:

All

Definition(s):

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

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 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.

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

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

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

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or MA9.

Reference(s):

1. NAPS Appendix R Report, Section 4.4 Attachment to Table 4-1
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 NOUE

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 (excluding Reactor Containment)

AND

The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Notes 1, 13)

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

Note 13: A Reactor Containment fire alarm is considered VALID upon receipt of multiple (more than one) fire zone alarms.

Table H-1 NAPS Fire Areas
• Cable Vaults & Tunnels
• Emergency Switchgear Rooms
• Emergency Diesel Generator Rooms
• Reactor Containment
• Quench Spray Pump Houses
• Safeguards Area
• Main Steam Valve House
• Cable Spreading Rooms
• Control Room
• CR Chiller Rooms
• Auxiliary / Fuel / Decontamination Buildings
• Fuel Oil Pump House Room A or B
• Service Water Pump House and Valve House
• Intake Structure Control House
• Auxiliary Service Water Pump House
• Auxiliary Feedwater Pump House
• Turbine Building

Mode Applicability:

All

Definition(s):

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

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 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).

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

With regard to Reactor Containment fire alarms, there is constant air movement in the enclosed containment due to the operation of the containment ventilation system. The operating cooling units are drawing air to the units past the smoke detectors. It can be reasonably expected that a fire that burns for 15 minutes would produce sufficient products of combustion to cause fire detectors in multiple zones to alarm. Therefore, a single Reactor Containment fire alarm is not considered VALID.

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

If an actual FIRE is verified by a report from the field, then HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

<u>Basis-Related Requirements from Appendix R (justification for the use of 30 minute criteria)</u>

10 CFR 50, Appendix R 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, 10 CFR 50, Appendix R, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in HU4.2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or MA9.

Reference(s):

1. NAPS Appendix R Report, Section 4.4 Attachment to Table 4-1
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	**NOUE**
A FIRE within the PLANT PROTECTED AREA or ISFSI Protected Area **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)	

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

Mode Applicability:

All

Definition(s):

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

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

Basis:

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

In addition to a FIRE addressed by EAL HU4.1 or HU4.2, a FIRE within the PLANT PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety.

This basis extends to a FIRE occurring within the Protected Area of an ISFSI located outside the PLANT PROTECTED AREA.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or MA9.

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	**NOUE**
A FIRE within the PLANT PROTECTED AREA or ISFSI Protected Area that requires an offsite fire department to assist with extinguishment	

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.

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

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

The Shift Fire Brigade Incident Commander will assess whether the fire conditions warrant outside assistance (ref. 1).

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or MA9.

Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gases

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

EAL:

HA5.1 Alert

Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 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**
Aux. Building El 274'	1, 2, 3, 4
Instrument Rack Rooms	4
Cable Vault & Tunnels	

Mode Applicability:

1 - Power Operation, 2 – Startup, 3 – Hot Standby, 4 - Hot Shutdown

Definition(s):

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

Basis:

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

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

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

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

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or IMPEDE a required action.
- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

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

Escalation of the emergency classification level would be via Category R, C or F ICs.

Reference(s):

1. Attachment 2, "Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases"
2. NEI 99-01 HA5

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations

EAL:

HA6.1	**Alert**
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel	

Mode Applicability:

All

Definition(s):

None

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Transfer of plant control begins when the last licensed operator leaves the Control Room.

Control will be established at the Auxiliary Shutdown Panel if the Control Room is evacuated for any reason (ref. 1, 2).

Escalation of the emergency classification level would be via IC HS6.

Reference(s):

1. 1(2)-AP-20, "Operation from the Auxiliary Shutdown Panel"
2. 0-FCA-1, "Control Room Fire"
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 has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel

AND

Control of **any** of the following key safety functions is **not** re-established within 15 min. of the last licensed operator leaving the Control Room (Note 1):

- Reactivity (modes 1, 2 and 3 **only**)
- Core cooling
- RCS heat removal

Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling

Definition(s):

None

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on SEM judgment. The SEM 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.

Control will be established at the Auxiliary Shutdown Panel if the Control Room was evacuated for any reason (ref. 1, 2).

Establishment of the reactivity safety function is only applicable in Modes 1, 2 and 3. Sufficient shutdown margin has already been established once in modes 4, 5 and 6 (ref.3).

Escalation of the emergency classification level would be via IC FG1 or CG1

Reference(s):

1. 1(2)-AP-20, "Operation from the Auxiliary Shutdown Panel"
2. 0-FCA-1, "Control Room Fire"
3. NRC EP FAQ 2015-014
4. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEM Judgment

Initiating Condition: Other conditions existing that in the judgment of the SEM warrant declaration of a NOUE

EAL:

HU7.1 NOUE

Other conditions exist which in the judgment of the SEM 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 SEM to fall under the emergency classification level description for a NOUE.

Reference(s):

1. NEI 99-01 HU7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEM Judgment

Initiating Condition: Other conditions exist that in the judgment of the SEM warrant declaration of an Alert

EAL:

HA7.1 **Alert**
Other conditions exist which, in the judgment of the SEM, 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 NAPS 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 NAPS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

OWNER CONTROLLED AREA - The entire area contiguous to the PLANT PROTECTED AREA, owned by the Company and designated to be controlled for security reasons.

PROJECTILE - An object directed toward a Nuclear Power Plant that could cause concern for its continued operability, reliability, or personnel safety.

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

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 SEM 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 – SEM Judgment

Initiating Condition: Other conditions existing that in the judgment of the SEM warrant declaration of a Site Area Emergency

EAL:

HS7.1 **Site Area Emergency**
Other conditions exist which in the judgment of the SEM 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 NAPS 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 NAPS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

OWNER CONTROLLED AREA - The entire area contiguous to the PLANT PROTECTED AREA, owned by the Company and designated to be controlled for security reasons.

PROJECTILE - An object directed toward a Nuclear Power Plant that could cause concern for its continued operability, reliability, or personnel safety.

SITE BOUNDARY - The power station proper and the 5000 ft radius circle from the center of the now abandoned Unit 3 containment.

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 SEM 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 – SEM Judgment

Initiating Condition: Other conditions exist that in the judgment of the SEM warrant declaration of a General Emergency

EAL:

HG7.1	**General Emergency**
Other conditions exist which in the judgment of the SEM 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.

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.

HOSTILE ACTION - An act toward NAPS 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 NAPS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

OWNER CONTROLLED AREA - The entire area contiguous to the PLANT PROTECTED AREA, owned by the Company and designated to be controlled for security reasons.

PROJECTILE - An object directed toward a Nuclear Power Plant that could cause concern for its continued operability, reliability, or personnel safety.

PLANT PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Plant Protected Area refers to the designated security area around the reactor and turbine buildings to which access is strictly controlled by the Plant Security Force.

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 SEM to fall under the emergency classification level description for a GENERAL EMERGENCY.

Reference(s):

1. NEI 99-01 HG7

Category M – System Malfunction

EAL Group: Hot Conditions (RCS temperature > 200°F); EALs in this category are applicable only in one or more hot operating modes.

Numerous system-related equipment failure events that warrant emergency classification have been identified in this category. They may pose actual or potential threats to plant safety.

The events of this category pertain to the following subcategories:

1. Loss of Emergency AC Power

Loss of emergency 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 4160V emergency 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 increase from these base-line levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

5. RCS Leakage

The reactor 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 trips. In the plant licensing basis, postulated failures of the RPS to complete a reactor trip 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 properly result in 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. Containment Failure

Failure of containment isolation capability (under conditions in which the containment is not currently challenged) warrants emergency classification. Failure of containment pressure control capability also warrants emergency classification.

9. Hazardous Event Affecting Safety Systems

Various natural and technological events that result in degraded plant safety system train performance or significant VISIBLE DAMAGE warrant emergency classification under this subcategory.

Category: M – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite AC power capability to emergency buses for 15 minutes or longer

EAL:

MU1.1 NOUE

Loss of **all** offsite AC power capability, Table M-1, to Unit 1(2) 4160V emergency buses H and J for ≥15 min. (Note 1)

Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.

Table M-1 AC Power Sources
Offsite: Unit 1 • Transfer Bus D • Transfer Bus F • Station Bus 1B • Station Bus 2B Unit 2 • Transfer Bus E • Transfer Bus F • Station Bus 2C • Station Bus 1A **Onsite:** • 1(2)H EDG • 1(2)J EDG • AAC (SBO) Diesel Generator (if already aligned)

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

Basis:

Table M-1 provides a list of offsite AC electrical power sources credited for this EAL.

Unit 1(2) 4160V emergency buses H and J are the emergency buses (ref. 1).

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC emergency buses. This condition represents a potential reduction in the level of safety of the plant.

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the emergency buses, whether or not the buses are powered from it.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of offsite power.

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down, the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs). The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). Additional bus ties for Unit 1 exist between the 1H emergency bus to 1B station service bus and 1J emergency bus to 2B station service bus which can provide a second independent offsite power sources to each Unit 1 emergency bus. Unit 2 emergency busses can be cross tied between the following: 2C station service bus to 2H and 1A station service bus to 2J, which can provide a second independent offsite power source to each Unit 2 emergency bus.

Escalation of the emergency classification level would be via IC MA1.

Reference(s):

1. 11715-FE-1A, "Main One Line Diagram (Unit 1)"
2. 12050-FE-1A, "Main One Line Diagram (Unit 2)"
3. 1(2)-ECA-0.0, "Loss of All AC Power"
4. 0-AP-10, "Loss of Electrical Power"
5. UFSAR Section 8.3
6. NEI 99-01 SU1

Category: M – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all but one** AC power source to emergency buses for 15 minutes or longer

EAL:

MA1.1 Alert

AC power capability, Table M-1, to Unit 1(2) 4160V emergency buses H and J 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 SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.

Table M-1 AC Power Sources
Offsite: Unit 1 • Transfer Bus D • Transfer Bus F • Station Bus 1B • Station Bus 2B Unit 2 • Transfer Bus E • Transfer Bus F • Station Bus 2C • Station Bus 1A **Onsite:** • 1(2)H EDG • 1(2)J EDG • AAC (SBO) Diesel Generator (if already aligned)

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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:

Table M-1 provides a list of offsite and onsite AC electrical power sources credited for this EAL.

Unit 1(2) 4160V emergency buses H and J are the emergency buses (ref. 1).

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC MU1.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an emergency bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from the unit main transformer.
- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being fed from an offsite power source.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down, the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs). The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). Additional bus ties for Unit 1 exist between the 1H emergency bus to 1B station service bus and 1J emergency bus to 2B station service bus which can provide a second independent offsite power sources to each Unit 1 emergency bus. Unit 2 emergency busses can be cross tied between the following: 2C station service bus to 2H and 1A station service bus to 2J, which can provide a second independent offsite power source to each Unit 2 emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions. A bus that is powered from the SBO can be credited as being powered from an independent power source. However, since it takes longer than 15 minutes to align the SBO diesel generator, the generator must be "already aligned" to credit it as an AC power source.

Escalation of the emergency classification level would be via IC MS1.

This hot condition EAL is equivalent to the cold condition EAL CU2.1.

Reference(s):

1. 11715-FE-1A, "Main One Line Diagram (Unit 1)"
2. 12050-FE-1A, "Main One Line Diagram (Unit 2)"
3. 1(2)-ECA-0.0, "Loss of All AC Power"
4. 0-AP-10, "Loss of Electrical Power"
5. UFSAR Section 8.3
6. NEI 99-01 SA1

Category: M – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite power and **all** onsite AC power to emergency buses for 15 minutes or longer

EAL:

MS1.1	**Site Area Emergency**
Loss of **all** offsite and **all** onsite AC power to Unit 1(2) 4160V emergency buses H and J for ≥15 min. (Note 1)	

Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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:

For this EAL credit can be taken for any AC power source that has sufficient capability to operate equipment necessary to maintain a safe shutdown condition, such as FLEX generators, provided it can be aligned within the 15 minute classification criteria.

Unit 1(2) 4160V emergency buses H and J are the emergency buses (ref. 1).

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A

breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down, the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs). The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). Additional bus ties for Unit 1exist between the 1H emergency bus to 1B station service bus and 1J emergency bus to 2B station service bus which can provide a second independent offsite power sources to each Unit 1 emergency bus. Unit 2 emergency busses can be cross tied between the following: 2C station service bus to 2H and 1A station service bus to 2J, which can provide a second independent offsite power source to each Unit 2 emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions. A bus that is powered from the SBO can be credited as being powered from an independent power source.

Escalation of the emergency classification level would be via ICs RG1, FG1 or MG1.

This hot condition EAL is equivalent to the cold condition EAL CA2.1.

Reference(s):

1. 11715-FE-1A, "Main One Line Diagram (Unit 1)"
2. 12050-FE-1A, "Main One Line Diagram (Unit 2)"
3. 1(2)-ECA-0.0, "Loss of All AC Power"
4. 0-AP-10, "Loss of Electrical Power"
5. UFSAR Section 8.3
6. NEI 99-01 SS1

Category: M –System Malfunction

Subcategory: 1 – Loss of Vital AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses

EAL:

MG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power to Unit 1(2) 4160V emergency buses H and J

AND

Core Cooling-RED Path conditions met

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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:

For this EAL credit can be taken for any AC power source that has sufficient capability to operate equipment necessary to maintain a safe shutdown condition, such as the FLEX generators.

This IC addresses a prolonged loss of all power sources to AC emergency buses that results in degraded core cooling. 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 eventually 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 threshold is based on meeting either CSFST Core Cooling Red Path criteria (ref. 6, 7):

- Core Exit Thermocouple readings ≥1,200 °F.

- Core exit TCs are ≥700°F with RCS subcooling based on core exit TCs ≤25°F [75°F], no RCPs are running, and RVLIS full range is ≤48%

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.

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down, the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs). The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). Additional bus ties for Unit 1 exist between the 1H emergency bus to 1B station service bus and 1J emergency bus to 2B station service bus which can provide a second independent offsite power sources to each Unit 1 emergency bus. Unit 2 emergency busses can be cross tied between the following: 2C station service bus to 2H and 1A station service bus to 2J, which can provide a second independent offsite power source to each Unit 2 emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions. A bus that is powered from the SBO can be credited as being powered from an independent power source.

Reference(s):

1. 11715-FE-1A, "Main One Line Diagram (Unit 1)"
2. 12050-FE-1A, "Main One Line Diagram (Unit 2)"
3. 1(2)-ECA-0.0, "Loss of All AC Power"
4. 0-AP-10, "Loss of Electrical Power"
5. UFSAR Section 8.3
6. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 2 Core Cooling"
7. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"
8. NEI 99-01 SG1

Category: M – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

Initiating Condition: Loss of **all** vital DC power for 15 minutes or longer

EAL:

MS2.1	**Site Area Emergency**
Indicated voltage is < 105 VDC on **all** vital 125 VDC battery buses for ≥15 min. (Note 1)	

Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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:

There are four independent 125 volt DC systems for each unit.

Each system consists of 125 volt DC distribution panels and its respective battery and a battery charger. The batteries 1(2)-I, 1(2)-II, 1(2)-III, and 1(2)-IV supply power only if the battery chargers fail or if the demand exceeds the capacity of the chargers. The batteries are rated for a minimum of two hours (ref. 1, 2).

A battery terminal voltage of 105 volts DC is the minimum voltage required to ensure proper operation of equipment connected to the DC bus (ref. 3).

This IC addresses a loss of vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via ICs RG1, FG1 or MG2.

This hot condition EAL equivalent of the cold condition EAL CU4.1.

Reference(s):

1. 1(2)-AP-10, "Loss of Electrical Power"
2. UFSAR Section 8.3.2, "Direct Current Power System"
3. 0-OP-6.4, "Operation of the SBO Diesel (SBO Event)"
4. NEI 99-01 SS8

Category: M –System Malfunction

Subcategory: 2 – Loss of Vital DC Power

Initiating Condition: Loss of **all** emergency AC and vital DC power sources for 15 minutes or longer

EAL:

MG2.1 General Emergency

Loss of **all** offsite and **all** onsite AC power to Unit 1(2) 4160V emergency buses H and J for ≥15 min. (Note 1)

AND

Indicated voltage is < 105 VDC on **all** vital 125 VDC battery buses for ≥15 min. (Note 1)

Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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 IC addresses a concurrent and prolonged loss of both emergency AC and vital DC power. A loss of all emergency AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both emergency AC and vital DC power will lead to multiple challenges to fission product barriers.

For this EAL credit can be taken for any AC power source that has sufficient capability to operate equipment necessary to maintain a safe shutdown condition, such as the FLEX generators.

The main generators are connected to the plant through the station service transformers (SSTs), which step the generator voltage down for distribution to the plant auxiliary systems. The generators are connected to the switchyard through the main transformers (MTs). A

breaker on the output of Unit 1 generator allows the generator to be electrically disconnected from the SSTs and MTs; the Unit 2 generator does not have a generator breaker. When a unit is shut down, the plant auxiliary systems are provided with electrical power from the switchyard through the MTs and SSTs or Reserve Station Service Transformers (RSSTs). The emergency buses are normally powered from the switchyard through redundant reserve station service transformers (RSSTs). Additional bus ties for Unit 1exist between the 1H emergency bus to 1B station service bus and 1J emergency bus to 2B station service bus which can provide a second independent offsite power sources to each Unit 1 emergency bus. Unit 2 emergency busses can be cross tied between the following: 2C station service bus to 2H and 1A station service bus to 2J, which can provide a second independent offsite power source to each Unit 2 emergency bus.

The station is equipped with an onsite blackout diesel generator that ensures a supply of power to at least one emergency 4160-Volt emergency bus during station blackout conditions when both emergency busses for a unit are initially lost. Under SBO conditions (for which the system was designed), the SBO diesel generator is used to supply power to one emergency bus on the unit which has initially lost both of its emergency busses. AP-10, Loss of Electrical Power, also allows the use of the SBO diesel generator to supply power to an emergency bus under non-blackout conditions. A bus that is powered from the SBO can be credited as being powered from an independent power source.

There are four independent 125 volt DC systems for each unit.

Each system consists of 125 volt DC distribution panels and its respective battery and a battery charger. The batteries 1(2)-I, 1(2)-II, 1(2)-III, and 1(2)-IV supply power only if the battery chargers fail or if the demand exceeds the capacity of the chargers. The batteries are rated for a minimum of two hours (ref. 4, 6).

A battery terminal voltage of 105 volts DC is the minimum voltage required to ensure proper operation of equipment connected to the DC bus (ref. 7).

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.

Reference(s):

1. 11715-FE-1A, "Main One Line Diagram (Unit 1)"
2. 12050-FE-1A, "Main One Line Diagram (Unit 2)"
3. 1(2)-ECA-0.0, "Loss of All AC Power"
4. 0-AP-10," Loss of Electrical Power"
5. UFSAR Section 8.3
6. UFSAR Section 8.3.2, "Direct Current Power System"
7. 0-OP-6.4, "Operation of the SBO Diesel (SBO Event)"
8. NEI 99-01 SG8

Category: M – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer

EAL:

MU3.1 **NOUE**
An UNPLANNED event results in the inability to monitor one or more Table M-2 parameters from within the Control Room for ≥15 min. (Note 1)

Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.

Table M-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core exit TC temperature • Level in at least one SG • Auxiliary feedwater flow to at least one SG

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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:

Applicable safety system parameters are listed in Table M-2.

The Plant Computer System/Safety Parameter Display System (SPDS) serve as redundant indicators which may be utilized as compensatory measures in lieu of the Control Room indicators associated with safety functions (ref. 1, 2).

The Inadequate Core Cooling Monitor (ICCM) System consists of three redundant subsystems that provide continuous control room displays: Core Exit Thermocouple (CET) System, Core Cooling Monitor (CCM) System, and Reactor Vessel Level Instrumentation System (RVLIS) (ref. 3).

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RCS 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 MA3.

Reference(s):

1. UFSAR Section 7.7.1.10, "Computer System"
2. UFSAR Section 7.8, "Emergency Response to Accidents"
3. UFSAR Section 7.9, "Inadequate Core Cooling Monitor (ICCM) System"

4. NEI 99-01 SU2

Category: M – 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:

MA3.1 Alert

An UNPLANNED event results in the inability to monitor one or more Table M-2 parameters from within the Control Room for ≥15 min. (Note 1)

AND

Any significant transient is in progress, Table M-3

Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.

Table M-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core exit TC temperature • Level in at least one SG • Auxiliary feedwater flow to at least one SG

Table M-3 Significant Transients
• Automatic turbine runback > 25% thermal reactor power • Electrical load rejection > 25% full electrical load • Reactor Trip • SI actuation

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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:

Applicable safety system parameters are listed in Table M-2.

Significant transients are listed in Table M-3.

The Plant Process Computer System/Safety Parameter Display System (SPDS) serve as redundant indicators which may be utilized as compensatory measures in lieu of the Control Room indicators associated with safety functions (ref. 1, 2).The Inadequate Core Cooling Monitor (ICCM) System consists of three redundant subsystems that provide continuous control room displays: Core Exit Thermocouple (CET) System, Core Cooling Monitor (CCM) System, and Reactor Vessel Level Instrumentation System (RVLIS) (ref. 3).

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an “inability to monitor” means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RCS water level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via ICs FS1 or RS1

Reference(s):

1. UFSAR Section 7.7.1.10, “Computer System “
2. UFSAR Section 7.8, “Emergency Response to Accidents”
3. UFSAR Section 7.9, “Inadequate Core Cooling Monitor (ICCM) System”
4. NEI 99-01 SA2

Category: M – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

MU4.1	**NOUE**
With letdown in service, Reactor Coolant Letdown Radiation Monitor 1(2)CH-RI-128(228) > 1.50E+04 mrem/hr	

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit (60 µCi/cc DEI-131) specified in Technical Specifications (ref. 1, 2). This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Per Engineering Calculation PA-0234, Rev. 1, the threshold value is indicative of more than 60 µCi/cc DEI-131 accident mix after 1 hour of decay. A monitor reading in excess of the threshold value 1.50E+04 mrem/hr (equivalent to 60 µCi/cc) indicates a challenge to the Technical Specification allowable limits for fuel clad degradation (ref. 1).

A portion of the letdown stream flows past radiation monitors 1(2)-CH-RI-128(228) to detect fission product activity in the reactor coolant and warn of a potential fuel element failure (ref. 3).

Escalation of the emergency classification level would be via IC FA1 or the Category R ICs.

Reference(s):

1. Calculation No. PA-0234, Rev. 1, "Post Accident Letdown Radiation Monitor Response for North Anna"
2. Technical Specifications 3.4.16, "RCS Specific Activity"
3. UFSAR Section 11.4.2.15, "Reactor Coolant Letdown Gross Activity Monitors"
4. NEI 99-01 SU3

Category: M – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

MU4.2 NOUE

Dose rate at 1 ft. from an unpressurized RCS sample ≥Table M-4

Table M-4 Tech. Spec. Coolant Activity Dose Rates	
Time > Shutdown (hrs)	**mR/hr/ml**
≤2	0.70
> 2 – ≤8	0.50
> 8	0.30

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Per Engineering Calculation RA-0059 (ref. 1), dose rate is assumed to result from radioactive iodines (I-131 thru I-135) in RCS in concentrations corresponding to 60 μCi/gm DEI-131. This value corresponds to the Technical Specification coolant activity limit for iodine spike at full power operations (ref. 2). The values contained in Table M-4 (Tech. Spec. Coolant Activity Dose Rates) represent expected one foot dose rates per ml of sample based on time since reactor shutdown to the time when the sample is taken. The expected dose rate is a near linear relationship with the volume of the sample, so any volume collected can be determined by dividing the measured dose rate by the sample volume and comparing to the threshold value from Table M-4 for the applicable time frame. These dose rates assume no emergency core cooling system (ECCS) injection so there is no dilution credited which would vary coolant volume. Values in the table have been rounded for ease of use. The > 8 hour threshold is conservative up to 24 hours following reactor shutdown. After 24 hours, the expected response from radioactive iodine levels off. Therefore, the value shown for > 8 hours applies for all samples taken 8 hours or more since reactor shutdown.

The values specified in Table M-4 were developed using a method to minimize error (+/-) for the threshold value within each defined time period. Values were chosen to minimize error from the highest to lowest dose rate within each range.

It should be noted that this EALs is primarily directed toward mechanical damage to the clad not involving inadequate core cooling (ICC) sequences. Clad damage due to ICC sequences is addressed by the fuel clad and CTMT fission product barrier thresholds (Category F).

Escalation of the emergency classification level would be via IC FA1 or the Category R ICs.

Reference(s):

1. RA-0059, "Detector Response to an RCS Sample for EAL Classification of Fuel Clad Degradation and Barrier Loss"
2. Technical Specifications 3.1.D
3. NEI 99-01 SU3

Category: M – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

MU4.3	**NOUE**

Sample analysis indicates that a reactor coolant activity value is > an allowable limit specified in Technical Specification 3.4.16

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Category R ICs.

Reference(s):

1. Technical Specifications 3.4.16, "RCS Specific Activity"
2. NEI 99-01 SU3

Category: M – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

MU5.1 NOUE

RCS unidentified or pressure boundary leakage > 10 gpm for ≥15 min.

OR

RCS identified leakage > 25 gpm for ≥15 min.

OR

Leakage from the RCS to a location outside containment > 25 gpm for ≥15 min.

(Note 1)

Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

UNISOLABLE - An open or breached system line that **cannot** be isolated, remotely or locally.

Basis:

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

Once the RCS leak rate has been quantified to be greater than the specified value, failure to isolate the leak within 15 minutes, or if known that the leak cannot be isolated within 15 minutes, from the time of leak rate quantification, requires immediate classification.

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

The first and second EAL conditions are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications) (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 containment, a secondary-side system (e.g., steam generator tube leakage) or a location outside of containment.

Unidentified leakage is all leakage (except RCP seal water injection or leak-off) that is not identified leakage. Pressure Boundary leakage is leakage (except SG leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall. Generally, leakage into *closed* systems, or leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the unidentified leakage monitoring systems or not to be from a fault in the reactor coolant pressure boundary, are called identified leakages.

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 (ref. 3, 4, 5).

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, locally or remotely).

Escalation of the emergency classification level would be via ICs of Category R or F.

Reference(s):

1. Technical Specification Section 1.1, "Definitions"
2. Technical Specification 3.4.13, "RCS Operational Leakage"
3. 1(2)-PT-52.2, "Reactor Coolant System Leak Rate (Hand Calculation)"
4. 1(2)-PT-52.2A, "Reactor Coolant System Leak Rate (Computer Calculation)"
5. 1(2)-AP-16, "Increasing Primary Plant Leakage"
6. NEI 99-01 SU4

Category: M – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

MU6.1 NOUE

An automatic trip did **not** shut down the reactor as indicated by reactor power ≥5% after **any** RPS setpoint is exceeded

AND

A subsequent automatic trip **OR** manual trip (trip switches or manual turbine trip) are successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

Note 8: A manual trip 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

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 failure of the RPS to initiate or complete an automatic reactor trip that results in a reactor shutdown (reactor power < 5%), and either a subsequent operator manual action taken at the reactor control consoles or an automatic trip (i.e., any subsequent RPS setpoint trip) is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor trip, operators will promptly initiate manual actions at the reactor control consoles to shut down the reactor (e.g., initiate a manual reactor trip using the reactor trip switches or manually tripping the main turbine). 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 (< 5%).

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 trip using the reactor trip switches or manually tripping the main turbine). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

A reactor trip resulting from actuation of the ATWS Mitigation System Actuation Circuitry (AMSAC) logic is considered a successful subsequent automatic reactor trip for the purposes of this EAL (ref. 3).

The plant response to the failure of an automatic trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC MA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC MA6 or FA1, an NOUE declaration is appropriate for this event.

A reactor shutdown is determined consistent with CSFST Subcriticality Red path criteria (ref. 1). Because the power level threshold for subcriticality RED path (5%) is the same as the Power Operation operating mode transition power, this EAL is only applicable in Mode 1.

Should a reactor trip signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip and the RPS fails to automatically shut down the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the trip failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

In the event that the operator identifies a reactor trip is IMMINENT and initiates a successful manual reactor trip before the automatic RPS trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. However, if subsequent manual reactor trip actions fail to shut down the reactor, the event escalates to the Alert under EAL MA6.1.

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 1 Subcriticality"
2. 1(2)-E-0, "Reactor Trip or Safety Injection"
3. UFSAR Section 7.2.1.1.6, "Turbine Trip-Reactor Trip"
4. NEI 99-01 SU5

Category: M – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

MU6.2 NOUE

A manual trip did **not** shut down the reactor as indicated by reactor power ≥5%

AND

A subsequent manual trip (trip switches or manual turbine trip) **OR** automatic trip is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

Note 8: A manual trip 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

Definition(s):

None

Basis:

This EAL addresses a failure of a manual reactor trip that results in a reactor shutdown (reactor power < 5%), and either a subsequent operator manual action taken at the reactor control consoles or an automatic trip is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems (< 5%) (ref. 1, 2).

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 trip using the reactor trip switches or manually tripping the main turbine). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

A reactor trip resulting from actuation of the ATWS Mitigation System Actuation Circuitry (AMSAC) logic is considered a successful subsequent automatic reactor trip for the purposes of this EAL (ref. 3).

The plant response to the failure of a manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC MA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC MA6 or FA1, an NOUE declaration is appropriate for this event.

A reactor shutdown is determined consistent with CSFST Subcriticality Red path criteria (ref. 1). Because the power level threshold for subcriticality RED path (5%) is the same as the Power Operation operating mode transition power, this EAL is only applicable in Mode 1.

Should a reactor trip signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip and the RPS fails to automatically shut down the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the trip failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 1 Subcriticality"
2. 1(2)-E-0, "Reactor Trip or Safety Injection"
3. UFSAR Section 7.2.1.1.6, "Turbine Trip-Reactor Trip"
4. NEI 99-01 SU5

Category: M – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Automatic or manual trip 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:

MA6.1 Alert

An automatic or manual trip did **not** shut down the reactor as indicated by reactor power ≥5%

AND

Subsequent automatic or manual trip actions (trip switches and manual turbine trip) are **not** successful in shutting down the reactor as indicated by reactor power ≥5% (Note 8)

Note 8: A manual trip 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

Definition(s):

None

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic reactor trip or failure of a manual reactor trip that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shut down the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

A manual action at the reactor control 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 trip using the reactor trip switches or manually tripping the main turbine). This action does not include locally tripping reactor trip and bypass breakers, 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 opening breakers). Actions taken at other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

A reactor trip resulting from actuation of the ATWS Mitigation System Actuation Circuitry (AMSAC) logic is considered a successful subsequent automatic reactor trip for the purposes of this EAL (ref. 3). Therefore an Alert classification would not be required.

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shut down the reactor is prolonged enough to cause a challenge to the core cooling or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC MS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC MS6 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.

A reactor shutdown is determined consistent with CSFST Subcriticality Red path criteria (ref. 1). Because the power level threshold for subcriticality RED path (5%) is the same as the Power Operation operating mode transition power, this EAL is only applicable in Mode 1.

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 1 Subcriticality"
2. 1(2)-E-0, "Reactor Trip or Safety Injection"
3. UFSAR Section 7.2.1.1.6, "Turbine Trip-Reactor Trip"
4. NEI 99-01 SA5

Category: M – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

EAL:

MS6.1 Site Area Emergency

An automatic or manual trip did **not** shut down the reactor as indicated by reactor power ≥5%

AND

All actions taken to shut down the reactor are **not** successful as indicated by reactor power ≥5%

AND EITHER:

- Core Cooling-RED Path conditions met
- Heat Sink-RED Path conditions met

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

This EAL addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

Reactor shutdown achieved by use of other trip actions such as locally opening supply breakers, emergency boration, or manually driving control rods are also credited as a successful manual trip if reactor power is < 5% before indications of an extreme challenge to either core cooling or heat removal exist (ref. 1, 2, 3).

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the 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 IC and EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shut down the reactor.

A reactor shutdown is determined consistent with CSFST Subcriticality Red path criteria (ref. 1).. Because the power level threshold for subcriticality RED path (5%) is the same as the Power Operation operating mode transition power, this EAL is only applicable in mode 1.

A severe challenge to adequate core cooling is based on meeting the Core Cooling Red path criteria (ref. 4, 5):

- Core Exit Thermocouple readings ≥1,200 °F.
- Core exit TCs are ≥700°F with RCS subcooling based on core exit TCs ≤25°F [75°F], no RCPs are running, and RVLIS full range is ≤48%.

The severe challenge to RCS heat removal is based on meeting the Heat Sink Red path criteria of both of the following conditions existing (ref. 6, 7):

- Narrow Range levels in all SGs < 11% [22%]
- Total feedwater flow to SGs ≤340 gpm

Escalation of the emergency classification level would be via IC RG1 or FG1.

Reference(s):

1. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 1 Subcriticality"
2. 1(2)-FR-S.1, "Response to Nuclear Power Generation / ATWS"
3. 1(2)-E-0, "Reactor Trip or Safety Injection"
4. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 2 Core Cooling"
5. 1(2)-FR-C.1, "Response to Inadequate Core Cooling"
6. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 3 Heat Sink"
7. 1(2)-FR-H.1, "Response to Loss of Secondary Heat Sink"
8. NEI 99-01 SS5

Category: M – System Malfunction

Subcategory: 7 – Loss of Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

MU7.1 NOUE

Loss of **all** Table M-5 onsite communication methods

OR

Loss of **all** Table M-5 State and local agency communication methods

OR

Loss of **all** Table M-5 NRC communication methods

Table M-5 Communication Methods

System	Onsite	State/ Local	NRC
Radio Communications System	X		
Public Address and Intercom System	X		
Private Branch Telephone Exchange (PBX)	X	X	X
Sound Powered Telephone System	X		
Commercial Telephone System		X	X
Automatic Ring Downs (SONET Ring)		X	
Instaphone Loop		X	
Dedicated NRC Communications			X

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to State and local agencies and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all State and local agencies of an emergency declaration. The State and local agencies referred to here are the Commonwealth of Virginia and local communities.

The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

This hot condition EAL is equivalent to the cold condition EAL CU5.1.

Reference(s):

1. North Anna Power Station Emergency Plan, Section 7.2, "Communications Systems"
2. UFSAR Section 7.7.1
3. NEI 99-01 SU6

Category: M – System Malfunction

Subcategory: 8 – Containment Failure

Initiating Condition: Failure to isolate containment or loss of containment pressure control

EAL:

MU8.1 NOUE

Any penetration is **not** closed within 15 min. of a VALID Phase A or B isolation signal

OR

CTMT pressure > 28 psia with < one full train of CTMT depressurization equipment (Note 11) operating per design for ≥15 min.

(Note 1)

Note 1: The SEM should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded.

Note 11: One full train of containment depressurization equipment consist of one Quench Spray (QS) System and one Recirculation Spray (RS) System from either train operating together.

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – Hot Shutdown

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 failure of one or more containment penetrations to automatically isolate (close) when required by an actuation signal (ref. 1). It also addresses an event that results in high containment pressure with a concurrent failure of containment pressure control systems. Absent challenges to another fission product barrier, either condition represents potential degradation of the level of safety of the plant.

For the first condition, the containment isolation signal (Phase A or B) must be generated as the result of an off-normal/accident condition (e.g., a safety injection or high containment pressure); a failure resulting from testing or maintenance does not warrant classification. The determination of containment and penetration status – isolated or not isolated – should be made in accordance with the appropriate criteria contained in the plant APs and EOPs. The 15-minute criterion is included to allow operators time to manually isolate the required penetrations, if possible (ref. 1).

The second condition addresses a condition where containment pressure is greater than the setpoint (28 psia) at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design (ref. 4, 5).

The Quench Spray (QS) System, operating in conjunction with the Recirculation Spray (RS) System, is designed to cool and depressurize the containment structure to less than 2.0 psig in one hour and sub-atmospheric pressure in less than 6 hours following a Design Basis Accident. The combination of required equipment can be obtained from using equipment on either emergency busses in order to meet the "one full train" requirement (ref. 2, 3).

The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment sprays) are either lost or performing in a degraded manner.

This event would escalate to a Site Area Emergency in accordance with IC FS1 if there were a concurrent loss or potential loss of either the Fuel Clad or RCS fission product barriers.

Reference(s):

1. UFSAR Section 6.2.4, "Containment Isolation System"
2. Technical Specifications Section B 3.6.6, "Quench Spray (QS) System"
3. Technical Specifications Section B 3.6.7, "Recirculation Spray (RS) System"
4. 1(2)-F-0, "Critical Safety Function Status Trees, Attachment 5 Containment"
5. 1(2)-FR-Z.1, "Response to High Containment Pressure"
6. NEI 99-01 SU7

Category: M – System Malfunction

Subcategory: 9 – Hazardous Event Affecting Safety Systems

Initiating Condition: Hazardous event affecting SAFETY SYSTEMS needed for the current operating mode

EAL:

MA9.1 Alert

The occurrence of **any** Table M-6 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 second train of the SAFETY SYSTEM needed for the current operating mode
- Event damage has resulted in VISIBLE DAMAGE to the second 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 M-6 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/SEM

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 - Hot Standby, 4 – 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.

Basis:

This IC 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 second 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 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.

An event affecting equipment common to two or more trains of a safety system (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the common equipment) should be classified as an Alert under this EAL, as appropriate to the plant mode. By affecting the functionality of multiple trains of a safety system, the loss of the common equipment effectively meets the two-train impact criteria that underlie the EALs and bases.

An event affecting a single-train safety system (i.e., there are indications of degraded performance and/or VISIBLE DAMAGE affecting the one train) would not be classified under this EAL because the two-train impact criteria that underlie the EALs and bases would not be met. If an event affects a single-train safety system, then the emergency classification should be made based on plant parameters/symptoms meeting the EALs for another IC. Depending upon the circumstances, classification may also occur based on SEM judgement.

An event that affects two trains of a safety system (e.g., one train has indications of degraded performance and the other VISIBLE DAMAGE) that also has one or more additional trains should be classified as an Alert under this EAL, as appropriate to the plant mode. This approach maintains consistency with the two-train impact criteria that underlie the EALs and bases and is warranted because the event was severe enough to affect the functionality of two trains of a safety system despite plant design criteria associated with system and system train separation and protection. Such an event may have caused other plant impacts that are not immediately apparent.

Escalation of the emergency classification level would be via IC FS1 or RS1.

This hot condition EAL is equivalent of the cold condition EAL CA6.1.

Reference(s):

1. EP FAQ 2016-002
2. NEI 99-01 SA9

Background

NEI 99-01, Rev. 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:

The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.

The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).

Further, as specified in IC HA5:

The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.

NAPS Table R-2 and H-2 Bases

A review of station operating procedures identified the following mode dependent in-plant actions and associated areas that are required for normal plant operation, cooldown or shutdown:

In-Plant Actions (NAPS)	Safe Shutdown Area	Modes
Chemistry to perform RCS isotopic analysis	AB El 274	1, 2
Ensure boron concentration for Cold Shutdown	AB El 274	3, 4
Sample RCS to place RHR in service	AB El 274	3, 4
I&C to perform PT-44.41	Instrument Rack Room	4
Place RHR in service per OP-14.1	AB EL 274' Cable Vault & Tunnels	4

Control Room ventilation systems have adequate engineered safety/design features in place to preclude a Control Room evacuation due to the external release of a hazardous gas (UFSAR Section 9.4.1 Main Control Room and Relay Rooms). Therefore, the Control Room is not included in this assessment or in Tables R-2/H-2.

Ref: OP-3.7, "Unit Shutdown from Mode 1 to Mode 5 for Refueling"

Table R-2 & H-2 Results

Table R-2/H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode**
Aux. Building El 274'	1, 2, 3, 4
Instrument Rack Rooms	4
Cable Vault & Tunnels	