ATTACHMENT 1

10 CFR 50.54(q)(5) Procedure Change Summary Analysis

10 CFR 50.54(q)(5) Procedure Change Summary Analysis

Procedure/Title

Exelon Generation Company, LLC (EGC) is submitting the following Emergency Plan Addendum revision for Calvert Cliffs Nuclear Power Plant (CCNPP):

- EP-AA-1011, Addendum 3, Revision 3, *"Calvert Cliffs Nuclear Power Plant Emergency Action Levels "*

Description of Procedure

EP-AA-1011, Addendum 3 describes the Emergency Action Levels (EALs) implemented at CCNPP for entering Emergency Classification Levels (ECLs).

Discussion of Changes

This revision to EP-AA-1011, Addendum 3 was implemented in support of the replacement of the Main Vent Stack Wide Range Noble Gas Monitors (WRNGMs) for CCNPP Unit 2. Similar changes were previously made for CCNPP Unit 1 in Revision 2 to EP-AA-1011, Addendum 3, which was submitted by letter dated March 8, 2017.

As noted above, CCNPP has replaced the WRNGMs for Unit 2. The new monitor design is such that the existing EAL thresholds for EALs RG1.1, RS1.1, RA1.1 and RU1.1 need to be revised. Computation No. EP-CALC-00004 Revision 0, "CCNPP Wide Range Noble Gas Monitor (WRNGM) Emergency Action Level Thresholds," has been developed to determine the threshold values for EALs RG1.1, RS1.1, RA1.1 and RU1.1, due to the replacement WRNGMs. The EAL thresholds are developed using NEI 99-01, Revision 5 methodology for the following initiating conditions: Notice of Unusual Event (UE), Alert (ALERT), Site Area Emergency (SAE), and General Emergency (GE). Detector responses and EAL thresholds are developed for the RD-52 low-range (1RE5416A, 1RE5416B, 2RE5416A, 2RE5416B), RD-72 mid-range (1RE5417A, 1RE5417B, 2RE5417A, 2RE5417B), and RD-72 high-range (1RE5418A, 1RE5418B, 2RE5418A, 2RE5418B) detectors from General Atomics that make up the WRNGMs. The RD-52 detector is sensitive to beta/electron emissions and the RD-72 detectors are sensitive to both beta/electron emissions and photon (gamma and x-ray) emissions. The WRNGMs under normal operation align the sample flow path to the RD-52 detector until a threshold limit programed into the microprocessor is exceeded. At this point, the WRNGMs switch the sample flow path over to the RD-72 detectors. The table below provides the EAL threshold values for each event classification.

Event Classification	EAL Threshold value
UE	5.19 E+05 μCi/sec
Alert	5.19 E+07 μCi/sec
SAE	5.19 E+08 μCi/sec
GE	5.19 E+09 μCi/sec

The WRNGMs are displayed in the Main Control Room as a single value on instrument RIC-5415.

The revised thresholds will be used in Table R-1 Effluent Monitor Classification Thresholds in EP-AA-1011, Addendum 3 and are applicable for CCNPP, Unit 2. The Unit 1 Table R-1 changes have already been incorporated into EP-AA-1011, Addendum 3.

In addition to the computation of the new threshold revisions, the basis section and basis reference section for EALs RG1.1, RS1.1, RA1.1, and RU1.1 in EP-AA-1011, Addendum 3, Attachment 1, "Emergency Action Level Technical Basis Plant-Specific," are being revised to remove information that pertained to the old Unit 2 WRNGMs and to reference Computation No. EP-CALC-00004 for the new thresholds for CCNPP, Units 1 and 2.

Description of How the Change Still Complies with Regulations

10 CFR Part 50, Appendix E, Section IV.B.1, requires: "*a means to determine the magnitude of and for continually assessing the impact of, the release of radioactive materials be described.*" 10 CFR 50.47(b)(9) requires that: "*Adequate methods, systems and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.*" Additionally, the guidance in NUREG-0654, Section II.I.2 stipulates: "*Onsite capability and resources to provide initial values and continuing assessment throughout the course of an accident shall include post-accident sampling capability, radiation and effluent monitors, in-plant iodine instrumentation, and containment radiation monitoring.*"

The "Plant-Specific" EAL Technical Basis was revised to reflect the replacement of the WRNGMs at CCNPP Unit 2. The design of the replacement monitors is such that the existing EAL threshold values for EALs RG1.1, RS1.1, RA1.1 and RU1.1 required revision. Computation No. EP-CALC-00004, Revision 0, *"CCNPP Wide Range Noble Gas Monitor (WRNGM) Emergency Action Level Thresholds,"* was developed to determine the threshold values for EALs RG1.1, RS1.1, RA1.1, and RU1.1 due to the replacement of the WRNGMs on CCNPP Unit 2. As noted above, similar changes were previously made for CCNPP Unit 1 in Revision 2 to EP-AA-1011, Addendum 3, which was submitted by letter dated March 8, 2017.

Updating the EAL threshold values based on an approved technical basis change does not alter the meaning or intent of the basis of the approved EALs. No emergency planning requirements have been deleted or minimized by this revision to EP-AA-1011, Addendum 3. The applicable emergency planning regulations and commitments to the NRC continue to be met.

Description of Why the Change is Not a Reduction in Effectiveness (RIE)

Updating the EAL threshold values based on an approved technical basis change does not alter the meaning or intent of the basis of the approved EALs. The applicable emergency planning regulatory requirements and commitments to the NRC continue to be satisfied. Therefore, the changes do not result in a reduction in effectiveness of the Emergency Plan for CCNPP.

ATTACHMENT 2

EP-AA-1011, Addendum 3, Revision 3, *"Calvert Cliffs Nuclear Power Plant Emergency Action Levels"*

Exelon Generation®

EP-AA-1011, Addendum 3
Revision 3

CALVERT CLIFFS

NUCLEAR POWER PLANT

EMERGENCY ACTION LEVELS

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ABBEVIATIONS / ACRONYMS

AC .... Alternating Current
APRM .... Average Power Range Meter
ATWS .... Anticipated Transient Without Scram
CAC .... Containment Air Cooler
CAS .... Central Alarm Station
CCW .... Component Cooling Water
CDE .... Committed Dose Equivalent
CE .... Combustion Engineering
CET .... Core exit Thermocouple
CFR .... Code of Federal Regulations
CHRRM .... Containment High Range Radiation Monitor
CR .... Control Room
CTMT/CNMT .... Containment
CSF .... Critical Safety Function
CSFST .... Critical Safety Function Status Tree
DC .... Direct Current
DEQ .... Dose Equivalent
DHR .... Decay Heat Removal
Disch .... Discharge
DOT .... Department of Transportation
DSC .... Dry Shielded Canister
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
ED .... Emergency Director
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
EPG .... Emergency Procedure Guideline
EPRI .... Electric Power Research Institute
ERG .... Emergency Response Guideline
EPIP .... Emergency Plan Implementing Procedure
ESF .... Engineered Safety Feature
ESW .... Emergency Service Water
FAA .... Federal Aviation Administration
FBI .... Federal Bureau of Investigation
FEMA .... Federal Emergency Management Agency

ACRONYMS & ABBREVIATIONS (continued)

GE....General Emergency
HOO....Headquarters (NRC) Operations Officer
HPSI....High Pressure Safety Injection
IC....Initiating Condition
IM....Instrument Maintenance
IPEEE....Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI....Independent Spent Fuel Storage Installation
Keff....Effective Neutron Multiplication Factor
LCO....Limiting Condition of Operation
LER....Licensee Event Report
LOCA....Loss of Coolant Accident
LPSI....Low Pressure Safety Injection
LWR....Light Water Reactor
MSIV....Main Steam Isolation Valve
MSL....Main Steam Line
mR....milliRoentgen
MW....Megawatt
MWS....Miscellaneous Waste System
NEI....Nuclear Energy Institute
NESP....National Environmental Studies Project
NPP....Nuclear Power Plant
NRC....Nuclear Regulatory Commission
NSSS....Nuclear Steam Supply System
NORAD....North American Aerospace Defense Command
NUMARC....Nuclear Management and Resources Council
OBE....Operating Basis Earthquake
OCA....Owner Controlled Area
ODCM/ODAM....Off-site Dose Calculation (Assessment) Manual
ORO....Off-site Response Organization
OTCC....Once Through Core Cooling
PA....Protected Area
PAG....Protective Action Guideline
POAH....Point of Adding Heat
PORV....Pressure Operated Relief Valve
PRA/PSA....Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR....Pressurized Water Reactor
PSIG....Pounds per Square Inch Gauge

ACRONYMS & ABBREVIATIONS (continued)

R .... Roentgen
RCC .... Reactor Control Console
RCIC .... Reactor Core Isolation Cooling
RCS .... Reactor Coolant System
rem .... Roentgen Equivalent Man
RETS .... Radiological Effluent Technical Specifications
RFP .... Refueling Pool Level
RPS .... Reactor Protection System
RPV .... Reactor Pressure Vessel
RVLIS .... Reactor Vessel Level Indicating System
RWCU .... Reactor Water Cleanup
RWST .... Reactor Water Storage Tank
SAE .... Site Area Emergency
SBO .... Station Blackout
SCBA .... Self Contained Breathing Apparatus
SG .... Steam Generator
SI .... Safety Injection
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
SSE .... Safe Shutdown Earthquake
SUR .... Startup Rate
TEDE .... Total Effective Dose Equivalent
TOAF .... Top of Active Fuel
TSC .... Technical Support Center
UE .... Unusual Event
UFSAR .... Updated Final Safety Analysis Report
WE .... Westinghouse Electric
WOG .... Westinghouse Owners Group
WRNGM .... Wide Range Noble Gas Monitor

1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Calvert Cliffs Nuclear Power Plant (CCNPP). The EAL Technical Bases Document is intended to provide clarification and understanding of how the EALs were developed for CCNPP as well as the intent of each EAL. This document ensures consistent understanding of the EAL scheme for decision makers. It should be used to facilitate review of the CCNPP EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EAL-TB-HOT or EAL-TB-COLD Emergency Action Level Matrix, may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Director in making classifications, particularly those involving judgment or multiple events. The basis information shall also be used in training, for explaining event classifications to off-site officials, and facilitates regulatory review and approval of the classification scheme.

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

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 CCNPP Emergency Plan.

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

NEI 99-01 (NUMARC/NESP-007) Revision 4 was subsequently issued for industry implementation. Enhancements over earlier revisions included:

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

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

Subsequently, Revision 5 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL FAQs. Using NEI 99-01 Revision 5 Final (February 2008), CCNPP conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

Many of the EALs derived from the NEI methodology are fission product barrier based. That is, the conditions that define the EALs are based upon 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. “Loss” means the barrier no longer assures Containment of radioactive materials; “potential loss” infers an increased probability of barrier loss and decreased certainty of maintaining the barrier.

The primary fission product barriers are:

A. Fuel Clad (FC): The fuel clad barrier consists of fuel bundle tubes composed of zirconium-based alloys that contain the fuel pellets.

B. Reactor Coolant System (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 (CNMT): 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.

2.3 Emergency Classification Based on Fission Product Barrier Degradation

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

Unusual Event:

Any loss or any potential loss of Containment

Alert:

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

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

2.4 EAL Relationship to EOPs and Critical Safety Function Status

Where possible, the EALs have been made consistent with and utilize the conditions defined in the CCNPP Emergency Operating Procedure (EOP) network. While the symptoms that drive operator actions specified in the EOPs are not indicative of all possible conditions which warrant emergency classification, they define the symptoms, independent of initiating events, for which reactor plant safety and/or fission product barrier integrity are threatened. When these symptoms are clearly representative of one of the NEI 99-01 Rev. 5 Initiating Conditions, they have been utilized as an EAL. This permits rapid classification of emergency situations based on plant conditions without the need for additional evaluation or event diagnosis. Although some of the EALs presented here are based on conditions defined in the EOPs, classification of emergencies using these EALs is not dependent upon EOP entry or execution. The EALs can be utilized independently or in conjunction with the EOPs.

2.5 Symptom-Based vs. Event-Based Approach

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

2.6 EAL Organization

The CCNPP EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
  - EALs applicable under all plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
  - EALs applicable only under hot operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, 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 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 CCNPP EAL categories/subcategories and their relationship to NEI 99-01 Rev. 5 Recognition Categories are listed below.

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
R – Abnormal **R**ad Release / Rad Effluent	1 – Offsite Rad Conditions 2 – Onsite Rad Conditions & Spent Fuel Events 3 – CR/CAS Rad
H – **H**azards and Other Conditions Affecting Plant Safety	1 – Natural or Destructive Phenomena 2 – Fire or Explosion 3 – Hazardous Gas 4 – Security 5 – Control Room Evacuation 6 – Judgment
E – ISFSI/DSC	None
Hot Conditions:	
S – **S**ystem Malfunction	1 – Loss of AC Power 2 – Loss of DC Power 3 – Criticality & RPS Failure 4 – Inability to Reach or Maintain Shutdown Conditions 5 – Instrumentation 6 – Communications 7 – Fuel Clad Degradation 8 – RCS Leakage
F – **F**ission Product Barrier Degradation	None
Cold Conditions:	
C – **C**old Shutdown / Refueling System Malfunction	1 – Loss of AC Power 2 – Loss of DC Power 3 – RCS Level 4 – RCS Temperature 5 – Communications 6 – Inadvertent Criticality

The primary tool for determining the emergency classification level is the EAL Classification Matrix. The user of the EAL Classification Matrix may (but is not required to) consult the EAL Technical Bases Document in order to obtain additional information concerning the EALs under classification consideration. The user should consult Sections 2.7 and 2.8, and Attachments 1 and 2 of this document for such information.

2.7 Technical Bases Information

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

<u>Category Letter & Title</u>

<u>Subcategory Number & Title</u>

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

Site-specific description of the generic IC given in NEI 99-01 Rev. 5.

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

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

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

    G = General Emergency

    S = Site Area Emergency

    A = Alert

    U = Unusual Event

3. Third character (number): Subcategory number within the given category. Subcategories are sequentially numbered beginning with the number one (1). If a category does not have a subcategory, this character is assigned the number one (1).
4. Fourth character (number): The numerical sequence of the EAL within the EAL subcategory. If the subcategory has only one EAL, it is given the number one (1).

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

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

EAL (enclosed in rectangle)

Wording of the EAL as it appears in the EAL Classification Matrix. Note that defined terms are presented in all uppercase letters within the EAL wording.

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 - Refuel, D - Defueled, or All. (See Section 2.8 for operating mode definitions.)

Basis:

A Generic basis section provides a description of the rationale for the EAL as provided in NEI 99-01 Rev. 5. This is followed by a Plant-Specific basis section that provides CCNPP-relevant information concerning the EAL. If the EAL wording contains a defined term, the definition of the term is included at the end of the plant-specific basis discussion.

CCNPP Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.8 Operating Mode Applicability (Technical Specifications Table 1.2)

1 Power Operations

Reactor shutdown margin is less than Technical Specification minimum required ($k_{eff} \geq$ 0.99) and greater than 5% rated thermal power.

2 Startup

Reactor shutdown margin is less than Technical Specification minimum required ($k_{eff} \geq 0.99$) and less than or equal to 5% rated thermal power.

3 Hot Standby

Reactor shutdown margin greater than Technical Specification minimum required ($k_{eff} < 0.99$) with coolant temperature (Tavg) greater than or equal to 300°F.

4 Hot Shutdown

Reactor shutdown margin greater than Technical Specification minimum required ($k_{eff} < 0.99$) with coolant temperature (Tavg) less than 300°F and greater than 200°F.

5 Cold Shutdown

Reactor shutdown margin greater than Technical Specification minimum required ($k_{eff} < 0.99$) with coolant temperature (Tavg) less than or equal to 200°F.

6 Refuel

Reactor vessel head is unbolted.

D Defueled

All reactor fuel removed from reactor pressure vessel (full core off load during refueling or extended outage).

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

2.9 Validation of Indications, Reports and Conditions

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

2.10 Planned vs. Unplanned Events

Planned evolutions involve preplanning to address the limitations imposed by the condition, the performance of required surveillance testing, and the implementation of specific controls prior to knowingly entering the condition in accordance with the specific requirements of the site's Technical Specifications. Activities which cause the site to operate beyond that allowed by the site's Technical Specifications, planned or unplanned, may result in an EAL threshold being met or exceeded. Planned evolutions to test, manipulate, repair, perform maintenance or modifications to systems and equipment that result in an EAL value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned and is within the operational limitations imposed by the specific operating license. However, these conditions may be subject to the reporting requirements of 10 CFR 50.72.

2.11 Classifying Transient Events

For some events, the condition may be corrected before a declaration has been made. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined, in other situations, further analyses may be necessary (e.g., coolant radiochemistry following an ATWS event, plant structural examination following an earthquake, etc.). Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met.

Existing guidance for classifying transient events addresses the period of time of event recognition and classification (15 minutes). However, in cases when EAL declaration criteria may be met momentarily during the normal expected response of the plant, declaration requirements should not be considered to be met when the conditions are a part of the designed plant response, or result from appropriate Operator actions.

There may be cases in which a plant condition that exceeded an EAL was not recognized at the time of occurrence but is identified well after the condition has occurred (e.g., as a result of routine log or record review), and the condition no longer exists. In these cases, an emergency should not be declared. Reporting requirements of 10 CFR 50.72 are applicable and the guidance of NUREG-1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, should be applied.

2.12 Multiple Simultaneous Events and Imminent EAL Thresholds

When multiple simultaneous events occur, the emergency classification level is based on the highest EAL reached. For example, two Alerts remain in the Alert category. Or, an Alert and a Site Area Emergency is a Site Area Emergency. Further guidance is provided in RIS 2007-02, Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events.

Since CCNPP is a multi-unit station with shared safety-related system and functions, emergency classification level upgrading must also consider the effects of a loss of a common system on more than one unit (e.g., potential for radioactive release from more than one core). For example, the control panels for both units are in close proximity within the same room. Thus, Control Room evacuation most likely would affect both units. There are a number of other systems and functions which may be shared. This must be considered in the emergency classification level declaration.

Although the majority of the EALs provide very specific thresholds, the Emergency Director (ED) must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the ED, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded. While this is particularly prudent at the higher emergency classes (the early classification may permit more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.

2.13 Emergency Classification Level Downgrading

Another important aspect of usable EAL guidance is the consideration of what to do when the risk posed by an emergency is clearly decreasing. A combination approach involving recovery from General Emergencies and some Site Area Emergencies and termination from Unusual Events, Alerts, and certain Site Area Emergencies causing no long term plant damage appears to be the best choice. Downgrading to lower emergency classification levels adds notifications but may have merit under certain circumstances.

3.0 REFERENCES

3.1 Developmental

3.1.1 NEI 99-01 Rev. 5 Final, Methodology for Development of Emergency Action Levels, February 2008, ADAMS Accession Number ML080450149

3.1.2 NRC Regulatory Issue Summary (RIS) 2003-18, Supplement 2, Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels Revision 4, Dated January 2003 (December 12, 2005)

3.1.3 RIS 2007-02 Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events

3.2 Implementing

3.2.1 EAL-HOT Matrix

3.2.2 EAL-COLD Matrix

3.2.3 EAL Comparison Matrix

3.2.4 EAL Matrix

3.3 Commitments

None

4.0 DEFINITIONS (ref. 3.1.1 except as noted)

Affecting Safe Shutdown

Event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable hot or cold shutdown condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in hot shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is not "affecting safe shutdown."

Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in cold shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is "affecting safe shutdown."

Airliner/Large Aircraft

Any size or type of aircraft with the potential for causing significant damage to the plant (refer to the Security Plan for a more detailed definition).

Bomb

Refers to an explosive device suspected of having sufficient force to damage plant systems or structures.

Civil Disturbance

A group of people violently protesting station operations or activities at the site.

Confinement Boundary

The barrier(s) between areas containing radioactive substances and the environment.

Containment Closure

The site specific procedurally defined actions taken to secure Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. As applied to CCNPP, Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components (SSC), as listed in STP O-55A, provide a functional barrier to fission product release.

Explosion

A rapid, violent, unconfined combustion, or catastrophic failure of pressurized/energized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

Extortion

An attempt to cause an action at the station by threat of force.

Faulted

In a steam generator, the existence of secondary side leakage that results in an uncontrolled drop in steam generator pressure or the steam generator being 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.

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 CCNPP 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 CCNPP. Non-terrorism-based EALs should be used to address such activities, (e.g., 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

Mitigation actions have been ineffective, additional actions are not expected to be successful, and trended information indicates that the event or condition will occur. Where imminent timeframes are specified, they shall apply.

Intrusion

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

Independent Spent Fuel Storage Installation (ISFSI)

A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

Normal Levels

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

Normal Plant Operations

Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from Normal Plant Operations.

Projectile

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

Protected Area

The site specific area which normally encompasses all controlled areas within the security Protected Area fence.

Ruptured

In a steam generator, existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

Sabotage

Deliberate damage, mis-alignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. Equipment found tampered with or damaged due to malicious mischief may not meet the definition of sabotage until this determination is made by security supervision.

Safety-Related Structures, Systems or Component (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.

Strike Action

Work stoppage within the Protected Area by a body of workers to enforce compliance with demands made on CCNPP. The strike action must threaten to interrupt Normal Plant Operations.

Unisolable

A breach or leak that cannot be promptly isolated.

Unplanned

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

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 equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

Vital Area

Any areas, normally within the CCNPP Protected Area, that contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

5.0 CCNPP-TO-NEI 99-01 EAL CROSSREFERENCE

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

CCNPP	NEI 99-01	
EAL	**IC**	**Example EAL**
RU1.1	AU1	1
RU1.2	AU1	3
RU2.1	AU2	1
RU2.2	AU2	2
RA1.1	AA1	1
RA1.2	AA1	3
RA2.1	AA2	2
RA2.2	AA2	1
RA3.1	AA3	1
RS1.1	AS1	1
RS1.2	AS1	2
RS1.3	AS1	4
RG1.1	AG1	1
RG1.2	AG1	2
RG1.3	AG1	4
EU1.1	E-HU1	1
CU1.1	CU3	1
CU2.1	CU7	1
CU3.1	CU1	1
CU3.2	CU2	1

CCNPP	NEI 99-01	
EAL	IC	Example EAL
CU3.3	CU2	2
CU4.1	CU4	1
CU4.2	CU4	2
CU5.1	CU6	1, 2
CU6.1	CU8	2
CA1.1	CA3	1
CA3.1	CA1	1, 2
CA4.1	CA4	1, 2
CS3.1	CS1	1
CS3.2	CS1	2
CS3.3	CS1	3
CG3.1	CG1	1
CG3.2	CG1	2
FU1.1	FU1	1
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1
HU1.2	HU1	2
HU1.3	HU1	3
HU1.4	HU1	4
HU1.5	HU1	5
HU2.1	HU2	1
HU2.2	HU2	2
HU3.1	HU3	1

CCNPP	NEI 99-01	
EAL	**IC**	**Example EAL**
HU3.2	HU3	2
HU4.1	HU4	1, 2, 3
HU6.1	HU5	1
HA1.1	HA1	1
HA1.2	HA1	2
HA1.3	HA1	3
HA1.4	HA1	4
HA1.5	HA1	6
HA1.6	HA1	5
HA2.1	HA2	1
HA3.1	HA3	1
HA4.1	HA4	1, 2
HA5.1	HA5	1
HA6.1	HA6	1
HS4.1	HS4	1
HS5.1	HS2	1
HS6.1	HS3	1
HG4.1	HG1	1
HG4.2	HG1	2
HG6.1	HG2	1
SU1.1	SU1	1
SU3.1	SU8	1
SU4.1	SU2	1
SU5.1	SU3	1
SU6.1	SU6	1, 2

CCNPP	NEI 99-01	
EAL	IC	Example EAL
SU7.1	SU4	2
SU7.2	SU4	1
SU8.1	SU5	1, 2
SA1.1	SA5	1
SA3.1	SA2	1
SA5.1	SA4	1
SS1.1	SS1	1
SS2.1	SS3	1
SS3.1	SS2	1
SS5.1	SS6	1
SG1.1	SG1	1
SG3.1	SG2	1

6.0 ATTACHMENTS

6.1 Attachment 1, Emergency Action Level Technical Bases

6.2 Attachment 2, Fission Product Barrier Loss / Potential Loss Matrix and Basis

Attachment 1

Category R – Abnormal Rad Levels / Rad Effluents

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. Offsite Rad Conditions

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. Onsite Rad Conditions & Spent Fuel Events

Sustained general area radiation levels in excess of those indicating loss of control of radioactive materials or those levels which may preclude access to vital plant areas also warrant emergency classification.

3. CR/CAS Rad

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

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment greater than 2 times the ODCM for 60 minutes or longer

EAL:

RU1.1 Unusual Event

ANY gaseous or liquid monitor reading > Table R-1 column "UE" for ≥ 60 min. (Note 2)

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SAE	Alert	UE
Gaseous				
WRNGM (RIC-5415)	5.19E+09µCi/sec	5.19E+08µCi/sec	5.19E+07µCi/sec	5.19E+05µCi/sec
Main Steam Effluent (RIC 5421/5422)	2.99E+04 µCi/cc	2.99E+03 µCi/cc	2.99E+02 µCi/cc	N/A
Main Vent (RI-5415)	N/A	N/A	N/A	2.0E+05 cpm
Waste Processing (RI-5410)	N/A	N/A	N/A	4.0E+05 cpm
Fuel Handling Area Vent (RI-5420)	N/A	N/A	N/A	3.4E+05 cpm
Liquid				
Liquid Waste Disch* (RE-2201)	N/A	N/A	N/A	8.4E+05 cpm

* with effluent discharge **not** isolated

Mode Applicability:

All

Basis:

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 2 x ODCM limit multiples are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged. For example, a release exceeding 4x ODCM for 30 minutes does not meet the threshold.

This EAL includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the IC.

This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

Plant-Specific

The main plant vents consist of the exhaust flow from the auxiliary building ventilation systems and the condenser offgas system. Batch releases from the Waste Gas Decay Tanks, Containment vents and Containment purges are also directed into this stream. Per ODCM Attachment 7, the Unit 1 and Unit 2 vent flow rates are assumed to be 56.3 m^3/sec and 42.7 m^3/sec, respectively. The values for the WRNGMs are determined by EP-CALC-00004, CCNPP Wide Range Noble Gas Monitor (WRNGM) Emergency Action Level Thresholds. Unit 1 and Unit 2 have a Geiger-Muller tube Main Vent Monitor (1-RI-5415 and 2-RI-5415) which is displayed in CPM. The EAL values were determined assuming annual average meteorology, RCS noble gas concentrations and dose conversion factors used for emergency preparedness offsite dose assessment. The total gaseous release corresponding to 2 times ODCM limits is approximately 0.114 mrem in one hour as calculated below.

1. 2 x ODCM = 2 x 500 mrem/year = 1000 mrem/year
2. Hours/year = 24 x 365 = 8760 hours/year
3. (1000 mrem/year) / (8760 hours/year) = 0.114 mrem/hour (or 1.14E-3 mSv/hour)

The values for the vent radiation monitor readings are based on 90% of the 2 maximum permissible concentration (MPC) as discussed in ODCM 3.11.2.1 at the site boundary. This reduction will account for events that may result in releases through both unit vents. The 10% factor allowance for the other unit vent is conservative because it is two to three orders of magnitude larger than the normal releases through each vent. For the main vent monitors, which read in CPM, the Unit 1 flow rate is assumed because it yields the lowest (most limiting) concentration.

ODCM limit corresponds to 1.8 E+5 μCi/sec (site total)

2 x 1.8 E+5 μCi/sec = 3.6 E+5 μCi/sec

Assume event in one unit, allow 10% for release from other unit

Unusual Event noble gas release rate limit = 0.9 x 3.6 E+5 μCi/sec = 3.24 E+5 μCi/sec

Minimum Concentration Corresponding to RI-5415 Reading

$$\text{Concentration} = \frac{\text{Release rate } (\mu\text{Ci/sec})}{\text{Flow rate (cc/sec)}}$$

Unit 1 ODCM flow rates = 56.3 m^3/sec

$$\text{Unit 1 Concentration} = \frac{3.24\text{ E+5 }\mu\text{Ci/sec}}{56.3\text{ m}^3\text{/sec} \times 10^6\text{ cc/m}^3}$$

=5.7 E-3 μCi/cc

Unit 2 ODCM flow rates = 42.7 m^3/sec

$$\text{Unit 2 Concentration} = \frac{3.24\text{ E+5 }\mu\text{Ci/sec}}{42.7\text{ m}^3\text{/sec} \times 10^6\text{ cc/m}^3}$$

=7.6 E-3 μCi/cc

The calculated concentration result is converted to CPM so it can be read on RI-5415.

Isotope	RCS Conc.	% Total	Unit 1 Conc. (μCi/cc)	Unit 2 Conc. (μCi/cc)	Monitor Eff. ($CPM/10^{-6}$)	Unit 1 CPM	Unit 2 CPM
Kr-85	0.43	9.62	5.5 E-4	7.3 E-4	35	1.9 E+4	2.6 E+4
Kr-85m	0.16	3.58	2.1 E-4	2.7 E-4	55	1.2 E+4	1.5 E+4
Kr-87	0.15	3.36	1.9 E-4	2.6 E-4	218	4.1 E+4	5.6 E+4
Kr-88	0.28	6.36	3.7 E-4	4.8 E-4	289	1.1 E+5	1.4 E+5
Xe-133	2.6	58.17	3.3 E-3	4.4 E-3	1.87	6.2 E+3	8.3 E+3
Xe-135	0.85	19.01	1.1 E-3	1.4 E-3	70	7.7 E+4	1.0 E+5
Totals	4.47	100.00	5.7 E-3	7.6 E-3		2.7 E+5	3.5 E+5

The lower of the Unit 1 and Unit 2 values is conservatively rounded to 2.0E+5 CPM.
In a similar manner, values were determined for the Waste Processing Monitor (1-RI-5410 and 2-RI-5410) assuming noble gas distribution for Waste Gas Decay Tank rupture, average annual meteorology and a nominal waste processing ventilation flow of 23.4 m^3/sec (49,500 CFM). At 2 DAC at the site boundary, this corresponds to a reading of 4.0 E+5 CPM.

In a similar manner to that shown for RI-5415, values were determined for the Fuel Handling Area Vent Monitor (0-RI-5420) assuming only monitor response to noble gas released from a Fuel Handling Incident, average annual meteorology and a nominal fuel handling area ventilation flow of 15.1 m^3/sec (32,000 CFM). At 2 DAC at the site boundary, per Reference 2 this corresponds to a reading of 3.4 E+5 CPM.

Analysis was also performed for potential releases through Access Control Point and ECCS Pump (PP) Room. Per Reference 2, 2 DAC at the site boundary corresponds to monitor readings for each of these locations that are greater than 1E+6 CPM (i.e., off-scale high). The ECCS PP Room Monitors (1/2-RI-5406) and the Access Control Monitor (0-RI-5425) are not considered here because they will be offscale high at the Unusual Event emergency classification level. Therefore, these monitors provide no useful information for this EAL and are excluded from consideration.

Liquid effluent is monitored by the Liquid Waste Discharge Radiation Monitor (0-RE-2201). A high radiation alarm from this monitor results in a signal to close the Liquid Waste Discharge Valves. If these valves do not shut, the operators stop the pump being used for the discharge and shut the Liquid Waste RMS Outlet valve.

The Steam Generator Blowdown liquid effluent monitor upper range cannot detect releases associated with two times the ODCM limits and therefore would be assessed per EAL RU1.2 sample analysis.

The designation "N/A" in Table R-1 indicates that the listed instrument range is insufficient to indicate the specified value and therefore no value is used.

A radiation monitor reading is valid when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not valid for classification.

CCNPP Basis Reference(s):

1. Radioactivity Release Emergency Action Levels, J.B. McIlvaine, JSB Associates, Inc., September 1990
2. Off-Site Dose Calculation Manual (ODCM) for the Calvert Cliffs Nuclear Power Plant
3. AOP-6B Accidental Release of Radioactive Liquid Waste
4. UFSAR Section 11.1.2.1 Liquid Waste Processing System
5. NEI 99-01 AU1
6. EP-CALC-00004, CCNPP Wide Range Noble Gas Monitor (WRNGM) Emergency Action Level Thresholds

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment greater than 2 times the ODCM for 60 minutes or longer

EAL:

RU1.2	**Unusual Event**
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x ODCM limits for ≥ 60 min. (Note 2)	

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Mode Applicability:

All

Basis:

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 2 x ODCM limit multiples are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged. For example, a release exceeding 4x ODCM for 30 minutes does not meet the threshold.

This EAL includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

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

Plant-Specific

Releases in excess of two times the site Offsite Dose Calculation Manual (ODCM) (ref. 1) instantaneous limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times the ODCM limit for 30 minutes does not exceed this initiating condition. Further, the ED should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes.

Sample analyses are "confirmed" when the analytical results have been confirmed by Chemistry.

CCNPP Basis Reference(s):

1. Off-Site Dose Calculation Manual (ODCM) for the Calvert Cliffs Nuclear Power Plant
2. NEI 99-01 AU1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment greater than 200 times the ODCM for 15 minutes or longer

EAL:

RA1.1 Alert

ANY gaseous monitor reading > Table R-1 column "Alert" for ≥ 15 min. (Note 2)

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SAE	Alert	UE
Gaseous				
WRNGM (RIC-5415)	5.19E+09μCi/sec	5.19E+08μCi/sec	5.19E+07μCi/sec	5.19E+05μCi/sec
Main Steam Effluent (RIC 5421/5422)	2.99E+04 μCi/cc	2.99E+03 μCi/cc	2.99E+02 μCi/cc	N/A
Main Vent (RI-5415)	N/A	N/A	N/A	2.0E+05 cpm
Waste Processing (RI-5410)	N/A	N/A	N/A	4.0E+05 cpm
Fuel Handling Area Vent (RI-5420)	N/A	N/A	N/A	3.4E+05 cpm
Liquid				
Liquid Waste Disch* (RE-2201)	N/A	N/A	N/A	8.4E+05 cpm

* with effluent discharge **not** isolated

Mode Applicability: All

Basis:

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 200 x ODCM limit multiples are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the Releases should not be prorated or averaged. For example, a release exceeding 600 x ODCM for 5 minutes does not meet the threshold.

This EAL includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

Plant-Specific

A description of the applicable monitors and the methods used to calculate EAL values for the WRNGMs are determined by EP-CALC-00004, CCNPP Wide Range Noble Gas Monitor (WRNGM) Emergency Action Level Thresholds.

The ECCS PP Room Monitors (1/2-RI-5406) and the Access Control Monitor (0-RI-5425) are not considered here because they will be offscale high at the Unusual Event emergency classification level. At the Alert level, the readings on the main vent monitors (1/2-RI-5415), the Waste Processing Vent Monitors (1/2-RI-5410), and the Fuel Handling Area Vent Monitor (0-RI-5420) are well above the top of the instrument indicated range (1.0E+6 CPM). Therefore, these monitors provide no useful information for this EAL and are excluded from consideration.

The purpose of the Main Steam Effluent Radiation Monitor System is to monitor possible noble gas releases to the atmosphere from the main steam line through the atmospheric steam dump valves, the main steam safety relief valves, and the auxiliary feedwater steam turbine exhaust. The system includes two radiation monitors (1/2-RI-5421 and 1/2-RI-5422) per unit - one radiation monitor for each steam generator. The noble gas release rate of 3.2E+7 μCi/second (which corresponds to a whole body dose of 10 mrem in one hour at the site boundary) may also occur through release via main steam safety valve or atmospheric dump valve.

EP-CALC-00001, *Determination of Emergency Action Level Criteria for Calvert Cliffs Nuclear Power Plant Main Steam Line Radiation Monitors,* outlines, in detail, the calculation and methodology demonstrated below.

RA1.1 Threshold for RIC-5421, RIC-5422

RG1.1 Threshold Value = 2.99 E+4 μCi/cc (see section RG1.1)

GE Site Boundary Dose Threshold = 1000 mrem

Alert Site Boundary Dose Threshold = 10 mrem

$$\text{Main Steam Monitor Reading } (\mu\text{Ci/cc}) = \frac{\text{Alert Site Boundary Dose Threshold}}{\text{GE Site Boundary Dose Threshold}} \text{ X RG1.1 Threshold Value}$$

$$= \frac{10 \text{ mrem}}{1000 \text{ mrem}} \text{ X } 2.99 \text{ E} + 4 \ \mu\text{Ci/cc}$$

$$= \mathbf{2.99\ E{+}2\ \mu Ci/cc}$$

Based on the March 14, 1993 SG tube rupture event at Palo Verde Unit 2, the main steam effluent monitors (RI-5421, RI-5422) may read N^{16} immediately following SG tube rupture and prior to reactor trip. However, given the short half-life of N^{16}, this should clear within the first minute following reactor trip.

The designation "N/A" in Table R-1 indicates that the listed instrument range is insufficient to indicate the specified value and therefore no value is used.

A radiation monitor reading is valid when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not valid for classification.

CCNPP Basis Reference(s):

1. Radioactivity Release Emergency Action Levels, J.B. McIlvaine, JSB Associates, Inc., September 1990
2. Off-Site Dose Calculation Manual (ODCM) for the Calvert Cliffs Nuclear Power Plant
3. NEI 99-01 AA1
4. EP-CALC-00001, *Determination of Emergency Action Level Criteria for Calvert Cliffs Nuclear Power Plant Main Steam Line Radiation Monitors*
5. EP-CALC-00004, CCNPP Wide Range Noble Gas Monitor (WRNGM) Emergency Action Level Thresholds

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment greater than 200 times the ODCM for 15 minutes or longer

EAL:

RA1.2 Alert
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODCM limits for ≥ 15 min. (Note 2)

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Mode Applicability:

All

Basis:

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 200 x ODCM limit are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged. For example, a release exceeding 600 x ODCM for 5 minutes does not meet the threshold.

This EAL includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

This EAL addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage.

Plant-Specific

Confirmed sample analyses in excess of two hundred times the site Offsite Dose Calculation Manual (ODCM) limits that continue for 15 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. This event escalates from the Unusual Event by raising the magnitude of the release by a factor of 100 over the Unusual Event level (i.e., 200 times ODCM). Prorating the 500 mRem/yr basis of the 10 CFR 20 non-occupational MPC limits for both time (8766 hr/yr) and the 200 multiplier, the associated Exclusion Area Boundary dose rate would be approximately 10 mRem/hr. If sample analysis indicates the threshold is met and nothing is done within 15 minutes to effect a release reduction, the ED can conclude that the EAL threshold is met without second sample results.

Sample analyses are "confirmed" when the analytical results have been confirmed by Chemistry.

CCNPP Basis Reference(s):

1. Off-Site Dose Calculation Manual (ODCM) for Calvert Cliffs Nuclear Power Plant
2. NEI 99-01 AA1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RS1.1 Site Area Emergency

ANY radiation monitor reading > Table R-1 column "SAE" for ≥ 15 min. (Note 1)

- Do **not** delay declaration awaiting dose assessment results
- If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (see EAL RS1.2)

Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SAE	Alert	UE
Gaseous				
WRNGM (RIC-5415)	5.19E+09μCi/sec	5.19E+08μCi/sec	5.19E+07μCi/sec	5.19E+05μCi/sec
Main Steam Effluent (RIC 5421/5422)	2.99E+04 μCi/cc	2.99E+03 μCi/cc	2.99E+02 μCi/cc	N/A
Main Vent (RI-5415)	N/A	N/A	N/A	2.0E+05 cpm
Waste Processing (RI-5410)	N/A	N/A	N/A	4.0E+05 cpm
Fuel Handling Area Vent (RI-5420)	N/A	N/A	N/A	3.4E+05 cpm
Liquid				
Liquid Waste Disch* (RE-2201)	N/A	N/A	N/A	8.4E+05 cpm

* with effluent discharge **not** isolated

Mode Applicability:

All

Basis:

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

The site specific monitor list in Table R-1 includes effluent monitors on all potential release pathways.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is 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 should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

The Emergency Director (ED) should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

A description of the applicable monitors and the methods used to calculate EAL values for the WRNGMs are determined by EP-CALC-00004, CCNPP Wide Range Noble Gas Monitor (WRNGM) Emergency Action Level Thresholds.

The meteorology and source term (noble gases) used in determining the monitor readings in EAL RS1.1 are the same as those used for determining the monitor readings in EALs RU1.1 and RA1.1. Consistent use of these variables for all three EALs creates a protocol that maintains consistent intervals between the monitor readings for the four emergency classifications. The monitor readings developed with this protocol are intended to be used only in situations in which normal dose assessment cannot be accomplished. Normal dose assessment methods incorporate the actual accident meteorological variables and are the preferred methods for determining when this EAL is met.

In keeping with this protocol, the monitor response release coefficient for a steam generator tube rupture contaminated with RCS activity is used rather than the release coefficient for a steam generator tube rupture contaminated with gap activity. This choice of source terms may appear to be non-conservative. It is important to consider, however, that the EALs in other categories compensate for the apparent lack of conservatism. A loss or challenge to plant safety systems is a precursor EAL for a radiological release of this magnitude. Furthermore, normal dose assessment is the primary method for determining the offsite consequences and takes into account accident meteorology and source term.

SAE noble gas site release rate

Scale up from uncorrected Unusual Event noble gas release rate of 3.6 E+5 μCi/sec

$$\text{SAE Value} = \frac{100 \text{ mRem/hr} \times 3.6\text{ E+5 } \mu\text{Ci/sec}}{0.114 \text{ mRem/hr (or .00114 mSv/hr)}}$$

$= 3.2$ E+8 μCi/sec

This value corresponds to a concentration of about 5 μCi/cc and falls well within the range of the WRNGM.

The purpose of the Main Steam Effluent Radiation Monitor System is to monitor possible noble gas releases to the atmosphere from the main steam line through the atmospheric steam dump valves, the main steam safety relief valves and the auxiliary feedwater steam turbine exhaust. The system includes two radiation monitors (1/2-RI-5421 and 1/2-RI-5422) per unit - one radiation monitor for each steam generator.

The noble gas release rate of 3.2E+8 μCi/sec (which corresponds a whole body dose of 100 mrem in one hour at the site boundary) may also occur through release via the main steam safety valve or atmospheric dump valve.

EP-CALC-00001, *Determination of Emergency Action Level Criteria for Calvert Cliffs Nuclear Power Plant Main Steam Line Radiation Monitors,* outlines, in detail, the calculation and methodology demonstrated below.

RS1.1 Threshold for RIC-5421, RIC-5422

RG1.1 Threshold Value = 2.99 E+4 μCi/cc (see section RG1.1)

GE Site Boundary Dose Threshold = 1000 mrem

SAE Site Boundary Dose Threshold = 100 mrem

$$\text{Main Steam Monitor Reading } (\mu\text{Ci/cc}) = \frac{\text{SAE Site Boundary Dose Threshold}}{\text{GE Site Boundary Dose Threshold}} \text{ X RG1.1 Threshold Value}$$

$$= \frac{100 \text{ mrem}}{1000 \text{ mrem}} \text{ X } 2.99 \text{ E} + 4 \ \mu\text{Ci/cc}$$

= 2.99 E+3 μCi/cc

Based on the March 14, 1993 SG tube rupture event at Palo Verde Unit 2, the main steam effluent monitors (RI-5421, RI-5422) may read N^{16} immediately following SG tube rupture and prior to reactor trip. However, given the short half-life of N^{16}, this should clear within the first minute following reactor trip.

Dose assessment performed in accordance with ERPIP 822, Initial Dose Assessment Manual Calculation Methods, or EP-AA-110-202, CCNPP Dose Assessment, is based on actual meteorology but the monitor reading thresholds in this EAL are based on assumed meteorology. As a result, dose assessment may indicate that emergency classification is not warranted, even though the monitor reading threshold has been exceeded. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of dose assessments are available when emergency classification is being evaluated (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1.

The designation "N/A" in Table R-1 indicates that the listed instrument range is insufficient to indicate the specified value and therefore no value is used.

A radiation monitor reading is valid when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not valid for classification.

CCNPP Basis Reference(s):

1. Radioactivity Release Emergency Action Levels, J.B. McIlvaine, JSB Associates, Inc., September 1990
2. Off-Site Dose Calculation Manual (ODCM) for the Calvert Cliffs Nuclear Power Plant
3. ERPIP 822, Initial Dose Assessment Manual Calculation Methods
4. EP-AA-110-202, CCNPP Dose Assessment
5. NEI 99-01 AS1
6. EP-CALC-00001, *Determination of Emergency Action Level Criteria for Calvert Cliffs Nuclear Power Plant Main Steam Line Radiation Monitors*
7. EP-CALC-00004, CCNPP Wide Range Noble Gas Monitor (WRNGM) Emergency Action Level Thresholds

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RS1.2	**Site Area Emergency**
Dose assessment using actual meteorology indicates doses > 100 mRem TEDE or 500 mRem thyroid CDE at or beyond the site boundary	

Mode Applicability:

All

Basis:

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is 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 should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

The 100 mRem TEDE dose is set at 10% of the EPA PAG, while the 500 mRem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

The site boundary is depicted on CCNPP ODCM Attachment 18 "Environmental Monitoring Sites" (ref. 1). The site boundary is approximately a one-mile radius around the Protected Area. Per dose assessment methodology, the SB designated the Exclusion Area Boundary (EAB) is defined as 1150 meters (0.71 miles), which is the minimum distance to the SB. ERPIP-822 assumes a SB or EAB of 0.7 miles.

CCNPP Basis Reference(s):

1. Off-Site Dose Calculation Manual (ODCM) for the Calvert Cliffs Nuclear Power Plant
2. ERPIP 822, Initial Dose Assessment Manual Calculation Methods
3. NEI 99-01 AS1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RS1.3 Site Area Emergency

Field survey results indicate closed window dose rates > 100 mRem/hr expected to continue for ≥ 60 min. at or beyond the site boundary

OR

Analyses of field survey samples indicate thyroid CDE > 500 mRem for 1 hr of inhalation at or beyond the site boundary (Note 1)

Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

Mode Applicability:

All

Basis:

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is 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 should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

The site boundary is depicted on CCNPP ODCM Attachment 18 "Environmental Monitoring Sites" (ref. 1). The site boundary is approximately a one-mile radius around the Protected Area. Per dose assessment methodology, the SB designated the Exclusion Area Boundary (EAB) is defined as 1150 meters (0.71 miles), which is the minimum distance to the SB. ERPIP-822 assumes a SB or EAB of 0.7 miles.

CCNPP Basis Reference(s):

1. Off-Site Dose Calculation Manual (ODCM) for the Calvert Cliffs Nuclear Power Plant
2. NEI 99-01 AS1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RG1.1 General Emergency

ANY radiation monitor reading > Table R-1 column "GE" for ≥ 15 min. (Note 1)

- Do **not** delay declaration awaiting dose assessment results
- If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (see EAL RG1.2)

Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SAE	Alert	UE
Gaseous				
WRNGM (RIC-5415)	5.19E+09µCi/sec	5.19E+08µCi/sec	5.19E+07µCi/sec	5.19E+05µCi/sec
Main Steam Effluent (RIC 5421/5422)	2.99E+04 µCi/cc	2.99E+03 µCi/cc	2.99E+02 µCi/cc	N/A
Main Vent (RI-5415)	N/A	N/A	N/A	2.0E+05 cpm
Waste Processing (RI-5410)	N/A	N/A	N/A	4.0E+05 cpm
Fuel Handling Area Vent (RI-5420)	N/A	N/A	N/A	3.4E+05 cpm
Liquid				
Liquid Waste Disch* (RE-2201)	N/A	N/A	N/A	8.4E+05 cpm

* with effluent discharge **not** isolated

Mode Applicability:

All

Basis:

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

The monitor list in Table R-1 includes effluent monitors on all potential release pathways.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is 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 should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

A description of the applicable monitors and the methods used to calculate EAL values for the WRNGMs are determined by EP-CALC-00004, CCNPP Wide Range Noble Gas Monitor (WRNGM) Emergency Action Level Thresholds.

The meteorology and source term (noble gases) used in determining the monitor reading EAL in RG1.1 are the same as those used for determining the monitor reading EALs in EALs RU1.1, RA1.1 and RS1.1. Consistent use of these variables for all four EALs creates a protocol that maintains consistent intervals between the monitor readings for the four emergency classifications. The monitor readings developed with this protocol are intended to be used only in situations in which normal dose assessment cannot be accomplished. Normal dose assessment methods incorporate the actual accident meteorological variables and are the preferred methods for determining when this EAL is met.

In keeping with this protocol, the monitor response release coefficient for a steam generator tube rupture contaminated with RCS activity is used rather than the release coefficient for a steam generator tube rupture contaminated with gap activity. This choice of source terms may appear to be non-conservative. The EALs in other categories compensate for this apparent lack of conservatism. A loss or challenge to plant safety systems is a precursor EAL for a radiological release of this magnitude. Normal dose assessment is the primary method for determining the offsite consequences and takes into account the accident meteorology and source term.

GE noble gas site release rate

Scale up from uncorrected Unusual Event release rate of 3.6 E+5 µCi/sec

$$\text{GE Value} = \frac{1000 \text{ mRem/hr} \times 3.6 \text{ E+5 } \mu\text{Ci/sec}}{0.114 \text{ mRem/hr (or .00114 mSv/hr)}}$$

= 3.2 E+9 µCi/sec

This value corresponds to a concentration of about 50 µCi/cc and falls well within the range of the WRNGM.

The purpose of the Main Steam Effluent Radiation Monitor System is to monitor possible noble gas releases to the atmosphere from the main steam line through the atmospheric steam dump valves, the main steam safety relief valves, and the auxiliary feedwater steam turbine exhaust. The system includes two radiation monitors (1/2-RI-5421 and 1/2-RI-5422) per unit – one radiation monitor for each steam generator.

The noble gas release rate of 3.2 E+9 µCi/sec (which corresponds a whole body dose of 1000 mrem in one hour at the site boundary) may also occur through release via main steam safety valve or atmospheric dump valve.

EP-CALC-00001, *Determination of Emergency Action Level Criteria for Calvert Cliffs Nuclear Power Plant Main Steam Line Radiation Monitors,* outlines, in detail, the calculation and methodolodgy demonstrated below:

RG1.1 Threshold for RIC-5421, RIC-5422

Release Rate = 3.2 E+9 µCi/sec (see above)

Safety Valve Flow Rate = 2.44 E+6 cc/sec

$$\text{Release Concentration } (\mu\text{Ci/cc}) = \frac{\text{Release Rate}}{\text{Safety Valve Flow Rate}}$$

$$= \frac{3.2 \text{ E+9 } \mu\text{Ci/cc}}{2.44 \text{ E+6 cc/sec}}$$

= 1.31 E+3 µCi/cc

Detector response factors were applied to obtain a Main Steam Monitor reading of:

2.99 E+4 µCi/cc

Based on the March 14, 1993 SG tube rupture event at Palo Verde Unit 2, the main steam effluent monitors (RIC-5421, RIC-5422) may read N^{16} immediately following SG tube rupture and prior to reactor trip. However, given the short half-life of N^{16}, this should clear within the first minute following reactor trip.

Dose assessment performed in accordance with ERPIP 822, Initial Dose Assessment Manual Calculation Methods, or EP-AA-110-202, CCNPP Dose Assessment, is based on actual meteorology but the monitor reading thresholds in this EAL are based on assumed meteorology. As a result, dose assessment may indicate that emergency classification is not warranted, even though the monitor reading threshold has been exceeded. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of dose assessments are available when emergency classification is being evaluated (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1.

The designation "N/A" in Table R-1 indicates that the listed instrument range is insufficient to indicate the specified value and therefore no value is used.

A radiation monitor reading is valid when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not valid for classification.

CCNPP Basis Reference(s):

1. Radioactivity Release Emergency Action Levels, J.B. McIlvaine, JSB Associates, Inc., September 1990
2. Off-Site Dose Calculation Manual (ODCM) for the Calvert Cliffs Nuclear Power Plant
3. ERPIP 822, Initial Dose Assessment Manual Calculation Methods
4. EP-AA-110-202, CCNPP Dose Assessment
5. NEI 99-01 AG1
6. EP-CALC-00001, *Determination of Emergency Action Level Criteria for Calvert Cliffs Nuclear Power Plant Main Steam Line Radiation Monitors*
7. EP-CALC-00004, CCNPP Wide Range Noble Gas Monitor (WRNGM) Emergency Action Level Thresholds

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RG1.2	**General Emergency**
Dose assessment using actual meteorology indicates doses > 1,000 mRem TEDE or 5,000 mRem thyroid CDE at or beyond the site boundary	

Mode Applicability:

All

Basis:

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is 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 should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

The 1,000 mRem TEDE dose is set at 100% of the EPA PAG, while the 5,000 mRem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

The site boundary is depicted on CCNPP ODCM Attachment 18 "Environmental Monitoring Sites" (ref. 1). The site boundary is approximately a one-mile radius around the Protected Area. Per dose assessment methodology, the SB designated the Exclusion Area Boundary (EAB) is defined as 1150 meters (0.71 miles), which is the minimum distance to the SB. ERPIP-822 assumes a SB or EAB of 0.7 miles.

CCNPP Basis Reference(s):

1. Off-Site Dose Calculation Manual (ODCM) for the Calvert Cliffs Nuclear Power Plant
2. ERPIP 822, Initial Dose Assessment Manual Calculation Methods
3. NEI 99-01 AG1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RG1.3	**General Emergency**

Field survey results indicate closed window dose rates > 1,000 mRem/hr expected to continue for ≥ 60 min. at or beyond the site boundary

OR

Analyses of field survey samples indicate thyroid CDE > 5,000 mRem for 1 hr of inhalation at or beyond the site boundary (Note 1)

Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

Mode Applicability:

All

Basis:

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is 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 should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

Plant-Specific

The site boundary is depicted on CCNPP ODCM Attachment 18 "Environmental Monitoring Sites" (ref. 1). The site boundary is approximately a one-mile radius around the Protected Area. Per dose assessment methodology, the SB designated the Exclusion Area Boundary (EAB) is defined as 1150 meters (0.71 miles), which is the minimum distance to the SB. ERPIP-822 assumes a SB or EAB of 0.7 miles.

CCNPP Basis Reference(s):

1. Off-Site Dose Calculation Manual (ODCM) for the Calvert Cliffs Nuclear Power Plant
2. NEI 99-01 AG1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 2 – Onsite Rad Conditions & Spent Fuel Events

Initiating Condition: Unplanned rise in plant radiation levels

EAL:

RU2.1 Unusual Event

UNPLANNED water level drop in a reactor refueling pathway as indicated by **ANY** of the following (Note 3):

- Inability to restore and maintain SFP level > Technical Specification limit (65 ft 7 in)
- Inability to restore and maintain RFP level > Technical Specification limit (56 ft 8.5 in)
- Report of visual observation of an uncontrolled drop in water level in the RFP or SFP

AND

Area radiation monitor reading rise on **ANY** of the following:

- SFP Area RM-320 EL-69 (0RIC-7023 Channel 4)
- Spent Fuel Handling Machine (0RIC-7023 Channel 3)
- Unit 1/2 CNTMT EL-69 (RI-5316A/B/C/D)

Note 3: If loss of water level in the refueling pathway occurs while in Mode 5, 6 or D, consider classification under EALs CU3.1, CU3.2 or CU3.3

Mode Applicability:

All

Basis:

Generic

This EAL addresses increased radiation levels as a result of water level decreases above irradiated fuel or events that have resulted, or may result, in unplanned increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant.

The refueling pathway is a combination of cavities, tubes, canals and pools. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.

For refueling events where the water level drops below the RPV flange classification would be via EAL CU3.1, CU3.2 or CU3.3. This event escalates to an Alert per EAL RA2.1 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Table for events in operating modes 1-4.

Plant-Specific

The reactor refueling pool (RFP), spent fuel pool (SFP) and fuel transfer canal comprise the refueling pathway.

The Spent Fuel Pool (SFP) is normally filled with borated water to a level of 67 ft. The SFP is equipped with a level switch in each half that actuates a low level alarm at 66 ft 6 in. The minimum level per Technical Specifications is 65 feet, 8.5 inches (21 feet, 6 inches above the fuel seated in the SFP) (ref. 6). The phrase "... inability to restore and maintain...level >..." allows the operator to visually observe the low water level condition, if possible, and to attempt water level restoration instructions as long as water level remains above the top of irradiated fuel.

When the fuel transfer canal is directly connected to the spent fuel pool and refueling pool, there could exist the possibility of uncovering irradiated fuel in the fuel transfer canal. Therefore, this EAL is applicable to cold conditions in which irradiated fuel is being transferred to and from the Reactor Vessel and refueling pool. In hot conditions, the refueling pool is empty and this EAL would apply to irradiated fuel in the spent fuel pool.

Technical Specifications requires that refueling pool water level be maintained 23 ft above irradiated fuel seated in the Reactor Vessel when moving fuel.

While a radiation monitor could detect a rise in dose due to a drop in the water level, it might not be a reliable indication, in and of itself, of whether or not the fuel is uncovered. For example, the reading on an area radiation monitor located on the refueling bridge may rise due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Elevated radiation monitor indications will need to be combined with another indicator (or personnel report) of water loss.

This event escalates to an Alert if irradiated fuel outside the reactor vessel is uncovered.

Definitions:

Unplanned

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

CCNPP Basis Reference(s):

1. AOP-6E Loss of Refueling Pool Level
2. System Description No. 67/68 Spent Fuel Pool and Cooling System
3. Shutdown Safety Surveillance H (Fuel Movement Observation)
4. Technical Specifications Section 3.9.6 Refueling Pool Water Level
5. Technical Specifications Section 3.7.13 SFP Water Level
7. TS-05.01 Technical Specification Action Value Bases Document
8. NEI 99-01 AU2

Category: R – Radioactivity Release / Area Radiation

Subcategory: 2 – Onsite Rad Conditions & Spent Fuel Events

Initiating Condition: Unplanned rise in plant radiation levels

EAL:

RU2.2 **Unusual Event**
UNPLANNED area radiation readings increases by a factor of 1,000 over NORMAL LEVELS

Mode Applicability:

All

Basis:

Generic

This EAL addresses increased radiation levels as a result of water level decreases above irradiated fuel or events that have resulted, or may result, in unplanned increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant.

This EAL addresses increases in plant radiation levels that represent a loss of control of radioactive material resulting in a potential degradation in the level of safety of the plant.

This EAL excludes radiation level increases that result from planned activities such as use of radiographic sources and movement of radioactive waste materials. A specific list of ARMs is not required as it would restrict the applicability of the threshold. The intent is to identify loss of control of radioactive material in any monitored area.

Plant-Specific

Assessment of this EAL may be made with survey readings using portable instruments as well as installed radiation monitors.

Definitions:

Unplanned

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

Normal Levels

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

CCNPP Basis Reference(s):

1. NEI 99-01 AU2

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 2 – Onsite Rad Conditions & Spent Fuel Events

Initiating Condition: Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the Reactor Vessel

EAL:

RA2.1 Alert

Alarm on **ANY** of the following radiation monitors due to damage to irradiated fuel or loss of water level:

- Fuel Handling Area Vent (RI-5420)
- SFP Area RM-320 EL-69 (0RIC-7023 Channel 4)
- Spent Fuel Handling Machine (0RIC-7023 Channel 3)
- Unit 1/2 CNTMT EL-69 (RI-5316A/B/C/D)

Mode Applicability:

All

Basis:

Generic

This EAL addresses increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent an actual or substantial potential degradation in the level of safety of the plant.

This EAL addresses radiation monitor indications of fuel uncovery and/or fuel damage.

Increased ventilation monitor readings may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background at the ventilation monitor due to water level decrease may mask increased ventilation exhaust airborne activity and needs to be considered.

While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.

Escalation of this emergency classification level, if appropriate, would be based on RS1.1, RS1.2, RS1.3, RG1.1, RG1.2 or RG1.3.

Plant-Specific

This EAL is defined by the specific areas where irradiated fuel is located such as the refueling pool, reactor vessel, or spent fuel pool.

The bases for the area radiation high alarms and Containment radiation high alarms are a spent fuel handling accident and are, therefore, appropriate for this EAL. Elevated readings on ventilation monitors may also be indication of a radioactivity release from the fuel, confirming that damage has occurred. However, elevated background at the monitor due to water level lowering may mask elevated ventilation exhaust airborne activity and needs to be considered. However, while radiation monitors may detect a rise in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. The following are examples of events in which the monitor could be responding properly yet not signaling damage to or uncovery of irradiated fuel outside the reactor vessel:

- Transfer or relocation of a source stored in or near the fuel pool.
- A planned evolution such as removal of the reactor head.
- Movement of spent fuel with fuel rods that have been known to be leaking.

Interpretation of these EAL thresholds requires some understanding of the actual radiological conditions present in the vicinity of the monitors.

CCNPP Basis Reference(s):

1. 1(2)AOP-6E Loss of Refueling Pool Level
2. OP-07 Shutdown Operations, Section 6.9.B Checklist for Fuel Movement"
3. NEI 99-01 AA2

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 2 – Onsite Rad Conditions & Spent Fuel Events

Initiating Condition: Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the Reactor Vessel

EAL:

RA2.2 Alert

A water level drop in a reactor refueling pathway that will result in irradiated fuel becoming uncovered

Mode Applicability:

All

Basis:

Generic

This event represents a loss of control over radioactive material and represent an actual or substantial potential degradation in the level of safety of the plant.

Escalation of this emergency classification level, if appropriate, would be based on RS1.1, RS1.2, RS1.3, RG1.1, RG1.2 or RG1.3.

Plant-Specific

This EAL is defined by the specific areas where irradiated fuel is located such as the refueling pool, Reactor Vessel or Spent Fuel Pool.

There is no direct indication that water level in the Spent Fuel Pool or refueling pool has dropped to the level of the fuel other than visual observation. Since there is no level indicating system in the fuel transfer canal, visual observation of loss of water level would also be required. If available, video cameras may allow remote observation. Depending on available level indication, the declared threshold may need to be based on indications of makeup rate or lowering in Refueling Water Storage Tank level (ref. 1).

The movement of irradiated fuel assemblies within Containment requires a minimum water level of 23 ft above the Reactor Vessel flange and the top of spent fuel in the SFP. During refueling activities, this maintains sufficient water level in the refueling cavity, fuel transfer canal and SFP. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (ref. 2, 3).

Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment.

CCNPP Basis Reference(s):

1. 1(2) AOP-6E Loss of Refueling Pool Level
2. Technical Specifications Section 3.9.6 Refueling Pool Water Level
3. Technical Specifications Section 3.7.13 SFP Water Level
4. NEI 99-01 AA2

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 3 – CR/CAS Rad

Initiating Condition: Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions

EAL:

RA3.1 Alert

Dose rates > 15 mRem/hr in **ANY** of the following areas requiring continuous occupancy to maintain plant safety functions:

- Control Room
- CAS

Mode Applicability:

All

Basis:

Generic

This EAL addresses increased radiation levels that: impact continued operation in areas requiring continuous occupancy to maintain safe operation or to perform a safe shutdown.

The cause and/or magnitude of the increase in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EAL may be involved.

Areas requiring continuous occupancy include the Control Room and any other control stations that are staffed continuously, such as the security alarm station CAS.

Plant-Specific

The Control Room and Central Alarm Station (CAS) must be continuously occupied in all plant operating modes at CCNPP.

There is no area radiation monitoring system at CCNPP for the Control Room or CAS. Abnormal radiation levels may be initially detected by the Control Room vent supply radiation monitor, routine radiological surveys and abnormal responses from electronic personnel dosimeters worn by personnel occupying these spaces.

CCNPP Basis Reference(s):

1. System Description # 77/79 Radiation Monitoring System
2. UFSAR Section 11.2.2.5 Auxiliary Building Shielding
3. NEI 99-01 AA3

Category E – Independent Spent Fuel Storage Installation (ISFSI/DSC)

EAL Group: Not Applicable (the EAL in this category is applicable independent of plant operating mode)

An independent spent fuel storage installation (ISFSI) is a complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage. A significant amount of the radioactive material contained within a cask/canister must escape its packaging and enter the biosphere for there to be a significant environmental effect resulting from an accident involving the dry storage of spent nuclear fuel. Formal offsite planning is not required because the postulated worst-case accident involving an ISFSI has insignificant consequences to the public health and safety.

A Notification of Unusual Event is declared on the basis of the occurrence of an event of sufficient magnitude that a loaded cask confinement boundary is damaged or violated. This includes classification based on a loaded fuel storage cask/canister confinement boundary loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

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.

Minor surface damage that does not affect storage cask/canister boundary is excluded from the scope of these EALs.

Category: E – ISFSI/DSC

Subcategory: Not Applicable

Initiating Condition: Damage to a loaded cask confinement boundary

EAL:

EU1.1	**Unusual Event**
Damage to a loaded cask CONFINEMENT BOUNDARY	

Mode Applicability:

Not applicable

Basis:

Generic

An UE in this EAL is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask confinement boundary is damaged or violated. This includes classification based on a loaded fuel storage cask confinement boundary loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

This EAL addresses a dropped cask, a tipped over cask, EXPLOSION, PROJECTILE damage, FIRE damage or natural phenomena affecting a cask (e.g., seismic event, tornado, etc.).

Plant-Specific

The CCNPP ISFSI utilizes the NUHOMS dry spent fuel storage system.

This EAL addresses any condition which indicates a loss of a cask confinement boundary and thus a potential degradation in the level of safety of the ISFSI. The cask confinement boundary is considered the Dry Shielded Canister (DSC).

Definitions:

Confinement Boundary

The barrier(s) between areas containing radioactive substances and the environment.

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.

CCNPP Basis Reference(s):

1. Calvert Cliffs ISFSI USAR
2. NEI 99-01 E-HU1

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 - Refuel, D – Defueled).

The events of this category pertain to the following subcategories:

1. Loss of 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 the 4 kV safeguard buses.

2. Loss of 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 the 125 VDC buses.

3. RCS Level

Reactor Vessel or RCS water level is directly related to the status of adequate core cooling and, therefore, fuel clad integrity. RCS levels associated with Category C EALs are listed in Table C-5.

4. RCS Temperature

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

5. Communications

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

6. Inadvertent Criticality

Inadvertent criticalities pose potential personnel safety hazards as well being indicative of losses of reactivity control.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: AC power capability to 4kV vital buses reduced to a single power source for ≥15 min. such that **ANY** additional single failure would result in a complete loss of all 4kV vital bus power

EAL:

CU1.1 Unusual Event

AC power capability to 4kV vital buses 11(21) and 14(24) reduced to a single power source, Table C-1, for ≥ 15 min. (Note 4)

AND

ANY additional single power source failure will result in a complete loss of all 4kV vital bus power

Note 4: The ED 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.

Table C-1 AC Power Sources	
Onsite	• 1(2)A DG • 1(2)B DG • 0C DG , if aligned
Offsite	• 500kV transmission line 5051* • 500kV transmission line 5052* • 500kV transmission line 5072* • SMECO line , if aligned * A credited 500kV line must have an independent 13kV service transformer

Mode Applicability:

5 - Cold Shutdown, 6 - Refuel, D - Defueled

Basis:

Generic

The condition indicated by this EAL is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a complete loss of 4 kV vital bus AC power to one or both units. This condition could occur due to a loss of off-site power with a concurrent failure of all but one emergency generator to supply power to its emergency bus. The subsequent loss of this single power source would escalate the event to an Alert in accordance with EAL CA1.1.

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

Plant-Specific

CCNPP essential buses are 4kV vital buses 11(21) and 14(24). There are five offsite power sources available to these buses:

- Three 500kV transmission lines (Lines 5051, 5052, and 5072) supply offsite power to the 500kV switchyard via the transmission network.
- One 69kV/13kV Southern Maryland Electric Cooperative (SMECO) line may be manually connected to either 13kV bus and then to the 4kV vital buses. Under certain operational conditions, 13kV bus(es) may be receiving power from SMECO or may be quickly connected to the SMECO tie-line. The SMECO line is not used to carry loads for an operating unit and may provide power to no more than two 4kV vital buses simultaneously.
- If a fault affects only one unit, power may be obtained from the 500kV supply of the unaffected unit through a single 13kV transformer. This is considered an offsite AC power source available to the affected unit.

Based on operational experience, if the SMECO line or the 0C DG is not already aligned, these cannot be considered available/capable of supplying the bus due to the time it will take to align them. In any case, if this cannot be accomplished within 15 minutes, they are not available and an Unusual Event must be declared.

In-house power is fed back from the 500KV ring bus through 2 13kV transformers (designated P-13000-1 and P-13000-2). Normally P-13000-1 supplies all of Unit -1 (except 14 4kV bus) and 21 4kV bus. P-13000-2 supplies all of Unit-2 (except 21 4kV bus) and 14 4kV bus. Either P-13000 is capable of supplying all loads on both Units.

There are five onsite AC power sources:

- 1A DG for bus 11
- 1B DG for bus 14
- 2A DG for bus 21
- 2B DG for bus 24
- 0C DG may be aligned to any vital 4kV bus on either unit.

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If multiple sources fail to energize the unit safety-related buses within 15 minutes, an Unusual Event is declared under this EAL. The subsequent loss of the single remaining power source escalates the event to an Alert under EAL CA1.1.

CCNPP Basis Reference(s):

1. UFSAR Section 8 and Figure 8-1
2. Technical Specifications LCO 3.8.2 AC Sources-Shutdown
3. Technical Specifications LCO 3.8.10 Distribution Systems-Shutdown
4. OI-21A-1 1A Diesel Generator
5. OI-21A-2 2A Diesel Generator
6. OI-21B-1 1B Diesel Generator
7. OI-21B-2 2B Diesel Generator
8. OI-21C OC Diesel Generator
9. STP-O-90 AC Sources and On-site Power Distribution Systems 7 Day Operability Verification
10. AOP-7I Loss of 4kV, 480 Volt, or 208/120 Volt Instrument Bus Power
11. AOP-3F Loss of Off-site Power While in MODES 3, 4, 5, or 6
12. EOP-2 Loss of Off-site Power
13. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: Loss of **all** offsite and **all** onsite AC power to 4kV vital buses for ≥ 15 min.

EAL:

CA1.1 Alert

Loss of **all** offsite and **all** onsite AC power, Table C-1, to 4kV vital buses 11(21) and 14(24) for ≥ 15 min. (Note 4)

Note 4: The ED 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.

Table C-1 AC Power Sources	
Onsite	• 1(2)A DG • 1(2)B DG • 0C DG , if aligned
Offsite	• 500kV transmission line 5051 • 500kV transmission line 5052 • 500kV transmission line 5072 • SMECO line , if aligned * A credited 500kV line must have an independent 13kV service transformer

Mode Applicability:

5 - Cold Shutdown, 6 - Refuel, D - Defueled

Basis:

Generic

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal and the Ultimate Heat Sink.

The event can be classified as an Alert when in cold shutdown, refuel, or defueled mode because of the significantly reduced decay heat and lower temperature and pressure, increasing the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL.

Escalating to Site Area Emergency, if appropriate, is by EALs in Category R.

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

Plant-Specific

The CCNPP vital buses are 4kV buses 11(21) and 14(24). There are five offsite power sources available to these buses:

- Three 500kV transmission lines (Lines 5051, 5052, and 5072) supply offsite power to the 500kV switchyard via the transmission network.
- One 69kV/13kV Southern Maryland Electric Cooperative (SMECO) line may be manually connected to either vital 13kV bus and then to the 4kV vital buses. Under certain operational conditions, 13kV bus(es) may be receiving power from SMECO or may be quickly connected to the SMECO tie-line. The SMECO line is not used to carry loads for an operating unit and may provide power to no more than two 4kV vital buses simultaneously.
- If a fault affects only one unit, power may be obtained from the 500kV supply of the unaffected unit through a single 13 kV transformer. This is considered an offsite AC power source available to the affected unit.

In-house power is fed back from the 500kV ring bus through 2 13kV transformers (designated P-13000-1 and P-13000-2). Normally P-13000-1 supplies all of Unit -1 (except 14 4kV bus) and 21 4kV bus. P-13000-2 supplies all of Unit-2 (except 21 4kV bus) and 14 4kV bus. Either P-13000 is capable of supplying all loads on both Units.

There are five onsite AC power sources:

- 1A DG for bus 11
- 1B DG for bus 14
- 2A DG for bus 21
- 2B DG for bus 24
- 0C DG may be aligned to any vital 4kV bus on either unit.

Based on operational experience, if the SMECO line or the 0C DG is not already aligned, these cannot be considered available/capable of supplying the bus due to the time it will take to align them. In any case, if this cannot be accomplished within 15 minutes, they are not available and an Alert must be declared.

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of all AC power to vital buses. Even though an essential bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus then the bus should not be considered operable.

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If the 0C DG is available but is not powering a vital bus within 15 minutes, the EAL remains applicable.

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

CCNPP Basis Reference(s):

1. UFSAR Section 8 and Figure 8-1
2. Technical Specifications LCO 3.8.2 AC Sources-Shutdown
3. Technical Specifications LCO 3.8.10 Distribution Systems-Shutdown
4. OI-21A-1 1A Diesel Generator
5. OI-21A-2 2A Diesel Generator
6. OI-21B-1 1B Diesel Generator
7. OI-21B-2 2B Diesel Generator
8. OI-21C OC Diesel Generator
9. STP-O-90 AC Sources and On-site Power Distribution Systems 7 Day Operability Verification
10. AOP-7I Loss of 4kV, 480 Volt, or 208/120 Volt Instrument Bus Power
11. AOP-3F Loss of Off-site Power While in MODES 3, 4, 5, or 6
12. EOP-2 Loss of Off-site Power
13. EOP-7 Station Blackout
14. EOP-8 Functional Recovery
15. NEI 99-01 CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of DC Power

Initiating Condition: Loss of **required** DC power for ≥ 15 min.

EAL:

CU2.1	**Unusual Event**
< 105 VDC for ≥ 15 min. on the 125 VDC buses (11, 12, 21 or 22) that are required to monitor and control the removal of decay heat (Note 4)	

Note 4: The ED 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.

Mode Applicability:

5 - Cold Shutdown, 6 - Refuel

Basis:

Generic

The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations.

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

Plant-Specific

The 125 VDC vital system is divided into four independent and isolated channels. Each channel consists of one battery, two battery chargers, one DC bus, multiple DC unit control panels, and two inverters. Each inverter has an associated vital AC distribution panel board. Power to the DC bus, DC unit control panels, and inverters is supplied by the station batteries and/or the battery chargers. Each battery charger is fully rated and can recharge a discharged battery while at the same time supplying the steady state power requirements of the system. A reserve 125 VDC system for the plant is completely independent and isolated from all four separation groups, yet is capable of replacing any of the 125 VDC batteries. This system consists of one battery, one battery charger, and the associated DC switching equipment. Only the battery may be transferred for replacement duty.

The safety-related station batteries have been sized to carry their expected shutdown loads following a plant trip/LOCA and loss of offsite power or following a station blackout without battery terminal voltage falling below 105 volts. The fifteen-minute interval was selected as a threshold to exclude transient or momentary power losses. The loss of the 1A Diesel Generator 125 VDC bus 14 or 0C Diesel Generator bus 16 does not constitute an entry condition for this EAL.

Maintenance on a DC bus may be performed periodically during shutdown conditions. The "required" 125 VDC buses signifies the minimum allowed by Technical Specifications for the mode of operation (Refer to Technical Specification 3.8.10 for required 125V DC Buses). If loss of the "required" bus results in the inability to maintain cold shutdown, consideration should be given to escalation to an Alert under EAL CA4.1.

CCNPP Basis Reference(s):

1. UFSAR Section 8.4.3 and Figure 8.9
2. EOP-0 Post-Trip Immediate Actions
3. EOP-2 Loss of Off-Site Power, Section V
4. AOP-7J Loss of 120 Volt Vital AC or 125 Volt Vital DC Power
5. Technical Specifications Bases 3.8.10
6. NEI 99-01 CU7

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Level

Initiating Condition: RCS leakage

EAL:

CU3.1 Unusual Event

RCS leakage results in the inability to maintain or restore **EITHER** of the following for ≥15 min. (Note 4):

Pressurizer level > 101 in.

OR

RCS level within the target band established by procedure (when the level band was established below 101 in.)

Note 4: The ED 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

Mode Applicability:

5 - Cold Shutdown

Basis:

Generic

This EAL is considered to be a potential degradation of the level of safety of the plant. The inability to maintain or restore level is indicative of loss of RCS inventory.

Relief valve normal operation should be excluded from this EAL. However, a relief valve that operates and fails to close per design should be considered applicable to this EAL if the relief valve cannot be isolated.

Prolonged loss of RCS inventory may result in escalation to the Alert emergency classification level via either EAL CA4.1 or EAL CA3.1.

Plant-Specific

When pressurizer level drops to 101 in., pressurizer heaters are deenergized. This condition is signaled by annunciator 1C06-ALM Window E-35, PZR HTR CUTOFF.

In Cold Shutdown mode, pressurizer level may be intentionally lowered below the heater cutoff setpoint (e.g., in preparation to detension the reactor vessel head, etc.). For such evolutions, this EAL is applicable if RCS level cannot be restored and maintained within the prescribed target band specified by procedure.

CCNPP Basis Reference(s):

1. 1C06-ALM Window E-35, PZR HTR CUTOFF
2. AOP-2, Excessive Reactor Coolant Leakage
3. UFSAR 7.4.4
4. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Level

Initiating Condition: RCS Leakage

EAL:

CU3.2	**Unusual Event**

UNPLANNED RCS level drop below **EITHER** of the following for ≥ 15 min. (Note 4):

Reactor Vessel flange (44 ft) (when the level band was established above the flange)

OR

Target band (when the level band was established below the flange)

Note 4: The ED 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

Mode Applicability:

6 - Refuel

Basis:

Generic

This EAL is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water level below the Reactor Vessel flange are carefully planned and procedurally controlled. An unplanned event that results in water level decreasing below the Reactor Vessel flange, or below the planned RCS water level for the given evolution (if the planned RCS water level is already below the Reactor Vessel flange), warrants declaration of a UE due to the reduced RCS inventory that is available to keep the core covered.

The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame then it may indicate a more serious condition exists.

Continued loss of RCS Inventory will result in escalation to the Alert emergency classification level via either EAL CA4.1 or EAL CA3.1.

This EAL involves a decrease in RCS level below the top of the Reactor Vessel flange that continues for 15 minutes due to an unplanned event. This EAL is not applicable to decreases in flooded reactor cavity level, which is addressed by EAL RU2.1, until such time as the level decreases to the level of the vessel flange.

Plant-Specific

The Reactor Vessel flange level is at 44 ft (43.97 ft) Refueling Pool level (ref. 2). RCS elevations and level indication capabilities are illustrated in Attachment 15 of AOP-3B (ref. 2).

Figure C-1 illustrates the RCS levels associated with Category C EALs.

This EAL involves a lowering in RCS level below the top of the Reactor Vessel flange, or the inability to maintain water level above the intended level when level is being intentionally maintained below the flange, that continues for fifteen minutes due to an unplanned event. This EAL is not applicable to drops in flooded refueling pool level (covered by lowering spent fuel pool water level in EAL RU2.1) until such time as the level lowers to the level of the vessel flange. If level continues to lower and reaches the bottom of the RCS Hot Leg (35.58 ft), escalation to the Alert level under EAL CA3.1 would be appropriate. If the level lowering is accompanied by RCS heatup, escalation to the Alert level under EAL CA4.1 may also be appropriate.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refuel mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refuel mode, normal means of core temperature indication and 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. Reactor Vessel water level is normally monitored using the following instruments:

- Refueling Pool Level LI-4140
- RCS Level Narrow Range LI-4138
- RCS Level Wide Range LI-4139
- Local refueling level indicator (LG-4139 and tygon tubing)
- Reactor Vessel Level Monitoring System (RVLMS)

If RCS water level will be below the bottom of the Pressurizer (48.5 ft el.), IM installs and calibrates the Refueling Level Cart in the Control Room and places the RCS Wide Range Level Monitoring System in service (LI-4139). The Wide Range Level High/Low alarms (LAH/L-4139) and Narrow Range Level High/Low alarms (LAH/L-4138) are set above/below the target RCS level. As water level is changed, the alarms are reset every two feet. Table 1 of OP-7, Shutdown Operations, provides a cross-reference of Refueling Pool level and RVLMS alarm lights to various plant component elevations.

Definitions:

Unplanned

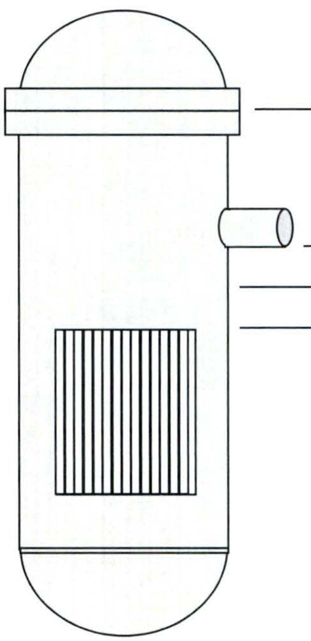
A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

CCNPP Basis Reference(s):

1. AOP-2, Excessive Reactor Coolant Leakage
2. AOP-3B, Abnormal Shutdown Cooling Conditions
3. OP-7 Shutdown Operations
4. NEI 99-01 CU2

Figure C-1: RCS Levels Thresholds (ref. 2)

Figure C-1 RCS Level Thresholds



	RVLMS	RCS	EAL
Reactor Vessel Flange	-----	44.0'	CU3.2
Bottom of Hotleg	29 in. alarm (6th)	35.6'	CA3.1
10 in. < Bottom of Hotleg	19 in. alarm (7th)	-----	CS3.1
Top of Active Fuel	10 in. alarm (8th)*	-----	CS3.2 CG3.1

* RVLMS last alarm actuates at 10" above Top of Active Fuel and should only be used if valid RCS level indication is not available

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Level

Initiating Condition: RCS Leakage

EAL:

CU3.3	**Unusual Event**
RCS level **cannot** be monitored with a loss of RCS inventory as indicated by an unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage	

Table C-2 RCS Leakage Indications
• Containment sump • Auxiliary Building sumps • Miscellaneous Waste System Tanks • RWT • RC Waste System Tank

Mode Applicability:

6 - Refuel

Basis:

Generic

This EAL is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water level below the Reactor Vessel flange are carefully planned and procedurally controlled. An UNPLANNED event that results in water level decreasing below the Reactor Vessel flange, or below the planned RCS water level for the given evolution (if the planned RCS water level is already below the Reactor Vessel flange), warrants declaration of a UE due to the reduced RCS inventory that is available to keep the core covered.

Continued loss of RCS Inventory will result in escalation to the Alert emergency classification level via either EAL CA3.1 or EAL CA4.1.

This EAL addresses conditions in the refueling mode when normal means of core temperature indication and RCS level indication may not be available. Redundant means of RCS 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. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RCS inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Containment to ensure they are indicative of RCS leakage.

Plant-Specific

In this EAL, all level indication would be unavailable and, the Reactor Vessel inventory loss must be detected by Containment sump, Auxiliary Building sumps, Miscellaneous Waste System Tanks, or RWT level changes. AOP-2A-1(2), Excessive Reactor Coolant Leakage, provides direction for determining RCS leakage for off normal events and for operations troubleshooting. Containment Sump narrow range level instrumentation (LI-4144/4145) on 1C08 (2C08) and 1C09 (2CO9) indicate level in the Containment Emergency Sump and can be trended prior to receiving the Containment Sump Level Hi alarm. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Containment to ensure they are indicative of RCS leakage.

Definitions:

Unisolable

A breach or leak that cannot be promptly isolated.

CCNPP Basis Reference(s):

1. AOP-2A Excessive Reactor Coolant Leakage
2. STP 0-27-1(2) RCS Leakage Evaluation
3. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Level

Initiating Condition: Loss of RCS inventory

EAL:

CA3.1	**Alert**

Loss of inventory as indicated by RCS water level < 35.6 ft (29 in. 6th alarm on RVLMS)

OR

RCS level **cannot** be monitored for ≥ 15 min. with a loss of RCS inventory as indicated by an unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage (Note 4)

Note 4: The ED 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.

Table C-2 RCS Leakage Indications
• Containment sump • Auxiliary Building sumps • Miscellaneous Waste System Tanks • RWT • RC Waste System Tank

Mode Applicability:

5 - Cold Shutdown, 6 - Refuel

Basis:

Generic

This EAL serves as a precursor to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS level decrease and potential core uncovery. This condition will result in a minimum emergency classification level of an Alert.

The inability to restore and maintain level after reaching this setpoint would be indicative of a failure of the RCS barrier.

If RCS level continues to lower then escalation to Site Area Emergency will be via EAL CS3.1, EAL CS3.2 or EAL CS3.3.

Plant-Specific

Figure C-1 illustrates the RCS levels associated with Category C EALs.

When RCS water level lowers to 35.58 ft (rounded to 35.6 ft), the bottom of the RCS hot leg is uncovered. This level can be monitored by:

- Refueling Pool Level LI-4140
- RCS Level Narrow Range LI-4138
- RCS Level Wide Range LI-4139
- Local refueling level indicator (LG-4139 and tygon tubing)
- RVLMS (6th RVLMS Alarm [29 in.]) corresponds to 35.58 ft

This EAL serves as a precursor to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates makeup systems have not been effective and may not be capable of preventing further RCS or Reactor Vessel level lowering and potential core uncovery. The bottom of the hot leg is the level equal to the bottom of the Reactor Vessel loop penetration, not the low point of the loop. This level was chosen because remote RCS level indication may be lost and loss of suction to decay heat removal systems has occurred. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier.

In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refuel mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refuel mode may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the fuel clad may be lower for events that occur in the Refuel mode with irradiated fuel in the Reactor Vessel. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refuel mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refuel mode, normal means of core temperature indication and 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.

In the second condition of this EAL, all level indication would be unavailable and, the Reactor Vessel/RCS inventory loss must be detected by Containment sump, Auxiliary Building sumps, Miscellaneous Waste System Tanks, or RWT level changes. AOP-2A-1(2), Excessive Reactor Coolant Leakage, provides direction for determining RCS leakage for off normal events and for operations troubleshooting. Containment Sump narrow range level instrumentation (LI-4144/4145) on 1C08(2C08) and 1C09(2CO9) indicate level in the Containment Emergency Sump and can be trended prior to receiving the Containment Sump Level Hi alarm. Sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Containment to ensure they are indicative of RCS leakage.

The 15-minute interval for the loss of level indication was chosen because it is half of the Site Area Emergency EAL duration. The interval allows this EAL to be an effective precursor to the Site Area Emergency EAL CS3.3. Therefore this EAL meets the definition for an Alert emergency.

Definitions:

Unisolable

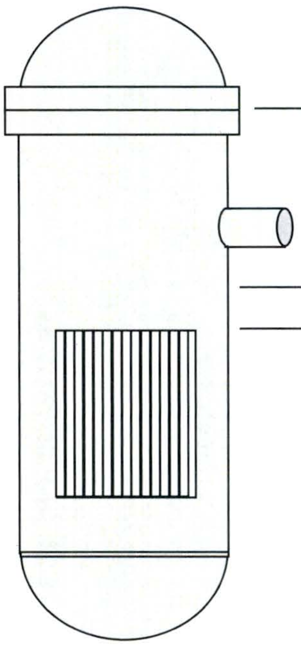
A breach or leak that cannot be promptly isolated.

CCNPP Basis Reference(s):

1. OP-7 Shutdown Operations
2. AOP-2A Excessive Reactor Coolant Leakage
3. AOP-3B, Abnormal Shutdown Cooling Conditions
4. STP 0-27-1(2) RCS Leakage Evaluation
5. NEI 99-01 CA1

Figure C-1: RCS Levels Thresholds (ref. 2)

Figure C-1 RCS Level Thresholds



	RVLMS	RCS	EAL
Reactor Vessel Flange	-----	44.0'	CU3.2
Bottom of Hotleg	29 in. alarm (6th)	35.6'	CA3.1
10 in. < Bottom of Hotleg	19 in. alarm (7th)	-----	CS3.1
Top of Active Fuel	10 in. alarm (8th)*	-----	CS3.2 CG3.1

* RVLMS last alarm actuates at 10" above Top of Active Fuel and should only be used if valid RCS level indication is not available

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Level

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

EAL:

CS3.1	**Site Area Emergency**
With CONTAINMENT CLOSURE **not** established, RCS level < 34.7 ft (19 in. 7th alarm on RVLMS)	

Mode Applicability:

5 - Cold Shutdown, 6 - Refuel

Basis:

Generic

Under the conditions specified by this EAL, continued decrease in RCS level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RCS. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via EAL CG3.1, EAL CG3.2, RG1.1, RG1.2 or RG1.3.

Plant-Specific

Figure C-1 illustrates the RCS levels associated with Category C EALs.

This level can be monitored by:

- Refueling Pool Level LI-4140
- RCS Level Narrow Range LI-4138
- RCS Level Wide Range LI-4139
- Local refueling level indicator (LG-4139 and tygon tubing)
- RVLMS (7th RVLMS Alarm) corresponds to 34.74 ft

When Reactor Vessel water level drops to 34.74 ft el. (rounded to 34.7 ft), level is ten inches below the bottom of the RCS hot leg vessel penetration.

Under the conditions specified in this EAL, continued lowering of RCS water level is indicative of a loss of inventory control. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the Reactor Vessel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS or Reactor Vessel water level lowering and core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and potential loss of the Fuel Clad barrier.

Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components, as listed in STP O-55A, provide a functional barrier to fission product release. Containment closure is initiated by the Shift Manager if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal. Containment closure requires that, upon a loss of decay heat removal, any open penetration must be closed or capable of being closed prior to RCS bulk boiling.

Definitions:

Containment Closure

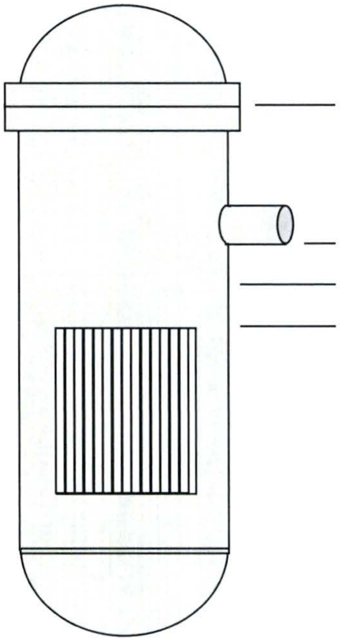
The site specific procedurally defined actions taken to secure Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. As applied to CCNPP, Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components (SSC), as listed in STP O-55A, provide a functional barrier to fission product release.

CCNPP Basis Reference(s):

1. UFSAR 7.5.9
2. OP-7 Shutdown Operations
3. AOP-3B, Abnormal Shutdown Cooling Conditions
4. ERPIP-601 Severe Accident Management Initial Diagnosis
5. NO-1-114 Containment Closure
6. STP O-55A-1(2) Containment Closure Verification
7. NEI 99-01 CS1

Figure C-1: RCS Levels Thresholds (ref. 2)

Figure C-1 RCS Level Thresholds



	RVLMS	RCS	EAL
Reactor Vessel Flange	-----	44.0'	CU3.2
Bottom of Hotleg	29 in. alarm (6th)	35.6'	CA3.1
10 in. < Bottom of Hotleg	19 in. alarm (7th)	-----	CS3.1
Top of Active Fuel	10 in. alarm (8th)*	-----	CS3.2 CG3.1
	* RVLMS last alarm actuates at 10" above Top of Active Fuel and should only be used if valid RCS level indication is not available		

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Level

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

EAL:

CS3.2	**Site Area Emergency**
With CONTAINMENT CLOSURE established, RCS level < 32.9 ft (10 in. alarm on RVLMS (Note 6))	

Note 6: The lowest RVLMS indication is the 10 in. alarm, which is 10 in. above top of active fuel. Therefore, this indicator should only be used when a valid RFP/RCS level indication is **not** available.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

Generic

Under the conditions specified by this EAL, continued decrease in RCS level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via EAL CG3.1, EAL CG3.2, RG1.1, RG1.2 or RG1.3.

Plant-Specific

This level can be monitored by:

- Refueling Pool Level LI-4140
- RCS Level Narrow Range LI-4138
- RCS Level Wide Range LI-4139
- Local refueling level indicator (LG-4139 and tygon tubing)
- RVLMS (7th RVLMS Alarm) corresponds to 34.74 ft

Figure C-1 illustrates the RCS levels associated with Category C EALs.

When Reactor Vessel/RCS water level drops to 32.9 ft el., core uncovery is about to occur. This level is below the lowest indicated hot leg level. The closest RVLMS indication is the 10 in. alarm. Therefore, this indicator should only be used when a valid RFP/RCS level indication is not available.

Under the conditions specified in this EAL, continued lowering of RCS water level is indicative of a loss of inventory control. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the Reactor Vessel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS or Reactor Vessel water level lowering and core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and potential loss of the Fuel Clad barrier.

Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components, as listed in STP O-55A, provide a functional barrier to fission product release. Containment closure is initiated by the Shift Manager if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal. Containment closure requires that, upon a loss of decay heat removal, any open penetration must be closed or capable of being closed prior to RCS bulk boiling.

Definitions:

Containment Closure

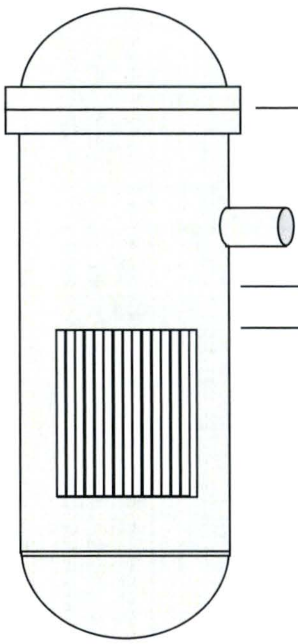
The site specific procedurally defined actions taken to secure Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. As applied to CCNPP, Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components (SSC), as listed in STP O-55A, provide a functional barrier to fission product release.

CCNPP Basis Reference(s):

1. UFSAR 7.5.9
2. OP-7 Shutdown Operations
3. AOP-3B, Abnormal Shutdown Cooling Conditions
4. ERPIP-601 Severe Accident Management Initial Diagnosis
5. NO-1-114 Containment Closure
6. STP O-55A-1(2) Containment Closure Verification
7. NEI 99-01 CS1

Figure C-1: RCS Levels Thresholds (ref. 2)

Figure C-1 RCS Level Thresholds



	RVLMS	RCS	EAL
Reactor Vessel Flange	-----	44.0'	CU3.2
Bottom of Hotleg	29 in. alarm (6th)	35.6'	CA3.1
10 in. < Bottom of Hotleg	19 in. alarm (7th)	-----	CS3.1
Top of Active Fuel	10 in. alarm (8th)*	-----	CS3.2 CG3.1

* RVLMS last alarm actuates at 10" above Top of Active Fuel and should only be used if valid RCS level indication is not available

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Level

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

EAL:

CS3.3 Site Area Emergency

RCS level **cannot** be monitored for ≥ 30 min.with a loss of RCS inventory as indicated by **ANY** of the following (Note 4):

- Containment radiation > 6 R/hr
- Erratic WRNI indication
- Unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage

Note 4: The ED 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.

Table C-2 RCS Leakage Indications
• Containment sump • Auxiliary Building sumps • Miscellaneous Waste System Tanks • RWT • RC Waste System Tank

Mode Applicability:

5 - Cold Shutdown, 6 - Refuel

Basis:

Generic

Under the conditions specified by this EAL, continued decrease in RCS level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RCS. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via EAL CG3.1, EAL CG3.2, RG1.1, RG1.2 or RG1.3.

The 30-minute duration allows sufficient time for actions to be performed to recover inventory control equipment.

As water level in the Reactor Vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in site specific monitor indication and possible alarm.

Plant-Specific

In Refuel or Cold Shutdown mode, normal RCS level indication (e.g., RVLMS) may be unavailable but alternate means of level indication are normally installed (including visual observation) to assure that the ability to monitor level will not be interrupted. If all means of level monitoring are not available, however, the Reactor Vessel inventory loss may be detected by the following indirect methods:

- 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 Containment radiation monitor indication and possible alarm. Containment radiation is indicated on 1(2)-RI-5317 A&B. Typical Containment radiation readings at full power are 1 to 1.2 R/hr. The Containment radiation monitors alarm at 6 R/hr. The 6 R/hr setpoint has been selected to be operationally significant and above that expected under normal plant conditions while in the Refuel mode.
- Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that source range monitors such as Wide Range Nuclear Instrumentation JI-001, -002, -003, -004 and the Shutdown Monitor can be used as a tool for making such determinations. Figure C-2 shows the response of the source range monitor during the first few hours of the TMI-2 accident. The instrument reported an increasing signal about 30 minutes into the accident. At this time, the reactor coolant pumps were running and the core was adequately cooled as indicated by the core outlet thermocouples. Hence, the increasing signal was the result of an increasing two-phase void fraction in the reactor core and vessel downcomer and the reduced shielding that the two-phase mixture provides to the source range monitor. Per CCNPP core damage assessment studies, core uncovery may be indicated when incore Rhodium neutron detectors or excore nuclear instruments indicate an output with the reactor known to be shutdown.
- If water level indication is unavailable, the RCS inventory loss may be detected by sump or tank level changes (Table C-2). Procedures provide instructions for calculating primary system leak rate by manual or computer-based water inventory balances. Sump/tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Containment to ensure they are indicative of RCS leakage.

Definitions:

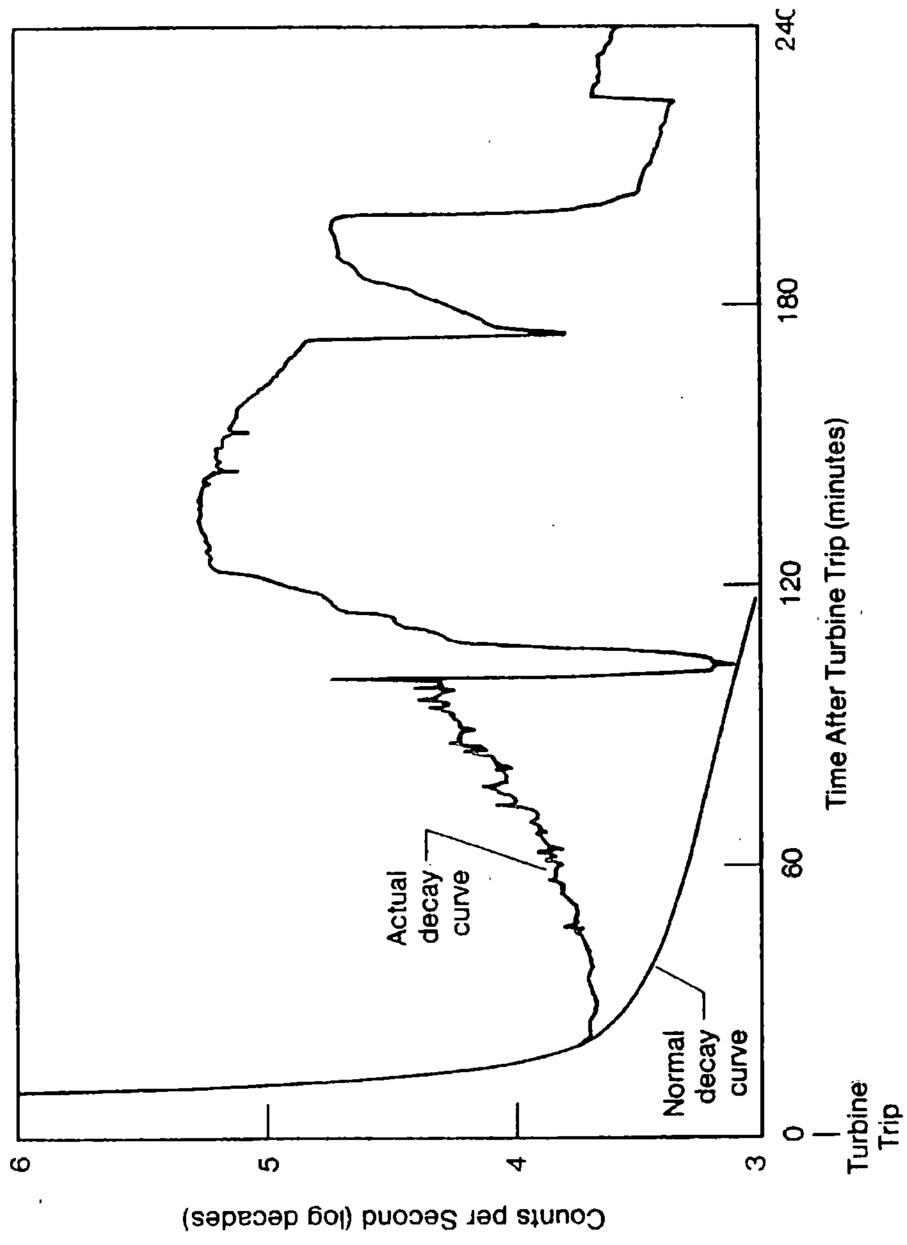
Unisolable

A breach or leak that cannot be promptly isolated.

CCNPP Basis Reference(s):

1. UFSAR 7.5.2
2. OI-35 Radiation Monitoring System
3. 1C10-ALM ESFAS 14 Alarm Manual, J-04
4. TS-76.01 RMS Area Radiation (Containment High Range) - Operable
5. ERPIP-601 Severe Accident Management Initial Diagnosis
6. ERPIP 602 Severe Accident Management Verification of Diagnosis
7. ERPIP-800 Core Damage Assessment
8. ERPIP-801 Core Damage Assessment Using Containment Radiation Dose Rates
9. OP-7 Shutdown Operations
10. AOP-2A Excessive Reactor Coolant Leakage
11. STP 0-27-1(2) RCS Leakage Evaluation
12. NEI 99-01 CS1

Figure C-2: Response of the TMI-2 Source Range Measurement During the First Six Hours of the Accident



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Level

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

EAL:

CG3.1	**General Emergency**
RCS level < 32.9 ft (10 in. alarm on RVLMS, Note 6) for ≥ 30 min. (Note 4)	
AND	
ANY Containment Challenge Indication, Table C-3	

Note 4: The ED 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.

Note 6: The lowest RVLMS indication is the 10 in. alarm, which is 10 in. above top of active fuel. Therefore, this indicator should only be used when a valid RFP/RCS level indication is **not** available.

Table C-3 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established • Hydrogen concentration in Containment ≥ 4% • UNPLANNED rise in Containment pressure

Mode Applicability:

5 - Cold Shutdown, 6 - Refuel

Basis:

Generic

This EAL represents the inability to restore and maintain RCS level to above the top of active fuel with Containment challenged. Fuel damage is probable if RCS level cannot be restored, as available decay heat will cause boiling, further reducing the RCS level. With the Containment breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a GE. The GE is declared on the occurrence of the loss or imminent loss of function of all three barriers.

A number of variables can have a significant impact on heat removal capability challenging the fuel clad barrier. Examples include: mid-loop, reduced level/flange level, head in place, cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining

Analysis indicates that core damage may occur within an hour following continued core uncovery therefore, 30 minutes was conservatively chosen.

If Containment Closure is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to General Emergency would not occur.

Plant-Specific

Figure C-1 illustrates the RCS levels associated with Category C EALs.

When Reactor Vessel water level drops to 32.90 ft el., core uncovery is about to occur. This level is below the lowest indicated hot leg level. The lowest RVLMS indication is the last alarm, which is at 10 in. above top of active fuel.

Under the conditions specified in this EAL, continued lowering of RCS water level is indicative of a loss of inventory control. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the Reactor Vessel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS or Reactor Vessel water level lowering and core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and potential loss of the Fuel Clad barrier. Fuel damage is probable if core submergence cannot be restored as available decay heat will cause boiling and further lowers the vessel level.

Three indications are associated with Containment challenges:

- Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components, as listed in STP O-55A, provide a functional barrier to fission product release. Containment closure is initiated by the Shift Manager if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal. Containment closure requires that, upon a loss of decay heat removal, any open penetration must be closed or capable of being closed prior to RCS bulk boiling.
- In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gases in Containment. However, Containment monitoring and/or sampling should be performed to verify this assumption. A combustible mixture can be formed when hydrogen gas concentration in the Containment atmosphere is greater than 4% by volume.

- Unplanned Containment pressure increases are not expected during Cold Shutdown or Refuel mode. The threshold is indicative of conditions challenging containment closure.

Definitions:

Containment Closure

The site specific procedurally defined actions taken to secure Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. As applied to CCNPP, Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components (SSC), as listed in STP O-55A, provide a functional barrier to fission product release.

Unplanned

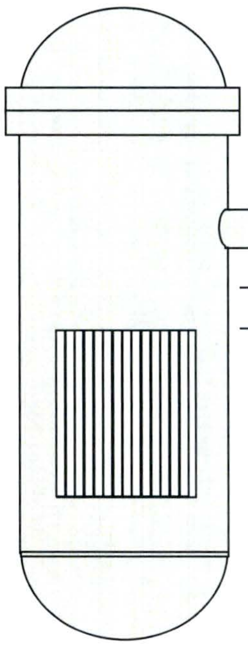
A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

CCNPP Basis Reference(s):

1. UFSAR 7.5.9
2. OP-7 Shutdown Operations
3. ERPIP-601 Severe Accident Management Initial Diagnosis
4. ERPIP-800 Core Damage Assessment
5. NO-1-114 Containment Closure
6. STP O-55A-1(2) Containment Closure Verification
7. UFSAR 7.5.8
8. Technical Specifications Table 3.3.10-1
9. OI-41A Hydrogen Recombiners
10. 1C10-ALM ESFAS 14 Alarm Manual, J-09
11. ERPIP-803 Core Damage Assessment Using Hydrogen
12. EOP-8 Functional Recovery Procedure
13. EOP-13.02 Hydrogen Concentration 4.0%
14. UFSAR 1.2.5
15. UFSAR 5.1.1
16. Operating License Amendment No. 242/DPR-53 & 216/DPR-69
17. NEI 99-01 CG1

Table C-1: RCS Levels Thresholds (ref. 2)

Figure C-1 RCS Level Thresholds



	RVLMS	RCS	EAL
Reactor Vessel Flange	-----	44.0'	CU3.2
Bottom of Hotleg	29 in. alarm (6th)	35.6'	CA3.1
10 in. < Bottom of Hotleg	19 in. alarm (7th)	-----	CS3.1
Top of Active Fuel	10 in. alarm (8th)*	-----	CS3.2 CG3.1

* RVLMS last alarm actuates at 10" above Top of Active Fuel and should only be used if valid RCS level indication is not available

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Level

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

EAL:

CG3.2 General Emergency

RCS level **cannot** be monitored with core uncovery indicated by **ANY** of the following for ≥ 30 min. (Note 4):

- Containment radiation > 6 R/hr
- Erratic WRNI indication
- Unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage

AND

ANY Containment Challenge Indication, Table C-3

Note 4: The ED 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

Table C-2 RCS Leakage Indications
• Containment sump • Auxiliary Building sumps • Miscellaneous Waste System Tanks • RWT • RC Waste System Tank

Table C-3 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established • Hydrogen concentration in Containment ≥ 4% • UNPLANNED rise in Containment pressure

Mode Applicability:

5 - Cold Shutdown, 6 - Refuel

Basis:

Generic

This EAL represents the inability to restore and maintain RCS level to above the top of active fuel with Containment challenged. Fuel damage is probable if RCS level cannot be restored, as available decay heat will cause boiling, further reducing the RCS level. With the Containment breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment: This is consistent with the definition of a GE. The GE is declared on the occurrence of the loss or imminent loss of function of all three barriers.

A number of variables can have a significant impact on heat removal capability challenging the fuel clad barrier. Examples include: mid-loop, reduced level/flange level, head in place, cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining

Analysis indicates that core damage may occur within an hour following continued core uncovery therefore, 30 minutes was conservatively chosen.

If Containment Closure is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to General Emergency would not occur.

Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Containment to ensure they are indicative of RCS leakage.

As water level in the RCS lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in site specific monitor indication and possible alarm.

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

Plant-Specific

In Refuel or Cold Shutdown mode, normal RCS level indication (e.g., RVLMS) may be unavailable but alternate means of level indication are normally installed (including visual observation) to assure that the ability to monitor level will not be interrupted. If all means of level monitoring are not available, however, the Reactor Vessel inventory loss may be detected by the following indirect methods:

- 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 Containment radiation monitor indication and possible alarm. Containment radiation is indicated on 1(2)-RI-5317 A&B. Typical Containment radiation readings at full power are 1 to 1.2 R/hr. The Containment radiation monitors alarm at 6 R/hr. The 6 R/hr setpoint has been selected to be operationally significant and above that expected under normal plant conditions while in the Refuel mode.

- Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that source range monitors such as Wide Range Nuclear Instrumentation JI-001, -002, -003, -004 and the Shutdown Monitor can be used as a tool for making such determinations. Figure C-2 shows the response of the source range monitor during the first few hours of the TMI-2 accident. The instrument reported an increasing signal about 30 minutes into the accident. At this time, the reactor coolant pumps were running and the core was adequately cooled as indicated by the core outlet thermocouples. Hence, the increasing signal was the result of an increasing two-phase void fraction in the reactor core and vessel downcomer and the reduced shielding that the two-phase mixture provides to the source range monitor. Per CCNPP core damage assessment studies, core uncovery may be indicated when incore Rhodium neutron detectors or excore nuclear instruments indicate an output with the reactor known to be shutdown.
- If water level indication is unavailable, the RCS inventory loss may be detected by sump or tank level changes (Table C-2). Procedures provide instructions for calculating primary system leak rate by manual or computer-based water inventory balances. Sump/tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Containment to ensure they are indicative of RCS leakage.

Three indications are associated with Containment challenges:

- Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components, as listed in STP O-55A, provide a functional barrier to fission product release. Containment closure is initiated by the Shift Manager if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal. Containment closure requires that, upon a loss of decay heat removal, any open penetration must be closed or capable of being closed prior to RCS bulk boiling.
- In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gases in Containment. However, Containment monitoring and/or sampling should be performed to verify this

assumption. A combustible mixture can be formed when hydrogen gas concentration in the Containment atmosphere is greater than 4% by volume.

- Unplanned Containment pressure increases are not expected during cold shutdown or refuel mode. The threshold is indicative of conditions challenging Containment closure.

Definitions:

Containment Closure

The site specific procedurally defined actions taken to secure Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. As applied to CCNPP, Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components (SSC), as listed in STP O-55A, provide a functional barrier to fission product release.

Unisolable

A breach or leak that cannot be promptly isolated.

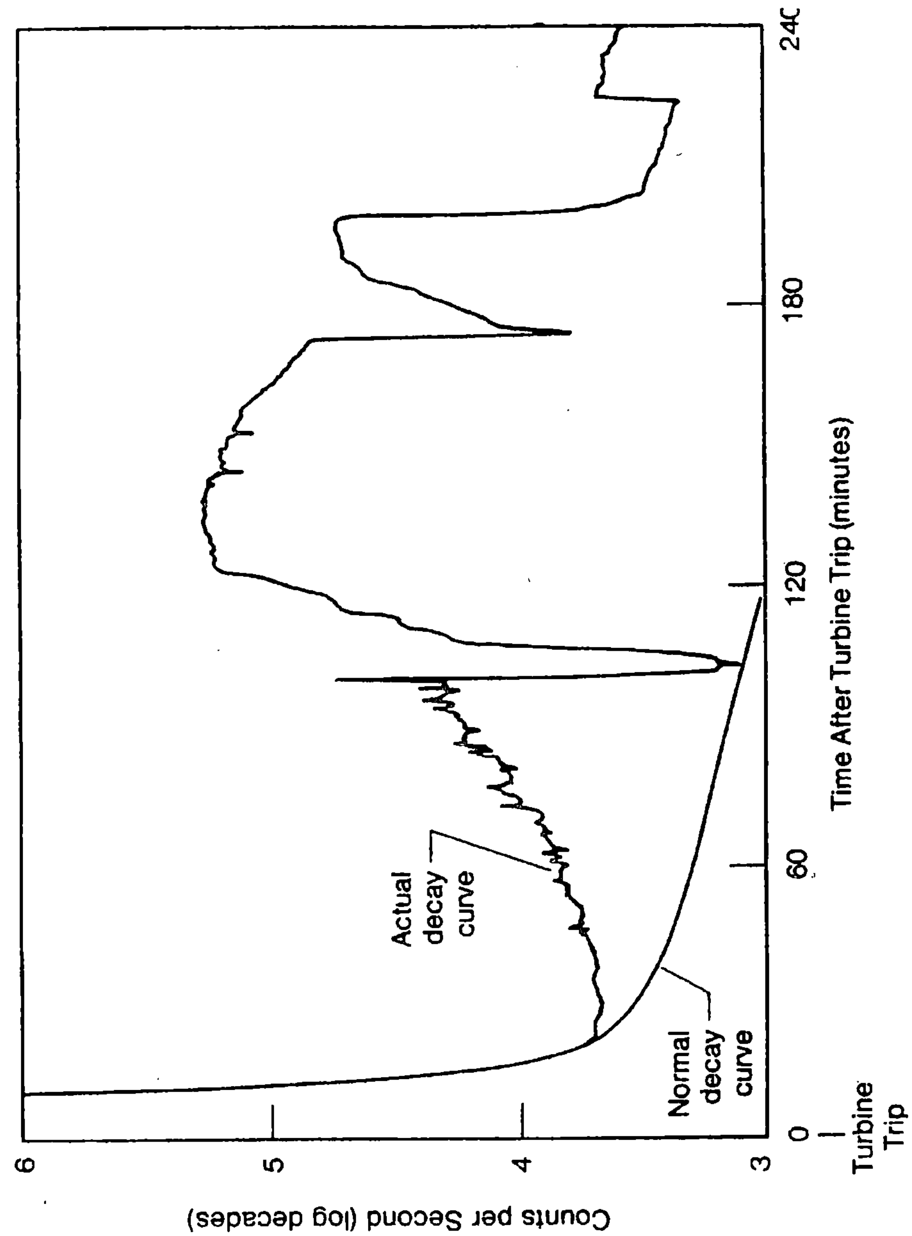
Unplanned

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

CCNPP Basis Reference(s):

1. UFSAR 7.5.9
2. ERPIP-601 Severe Accident Management Initial Diagnosis
3. ERPIP-800 Core Damage Assessment
4. NO-1-114 Containment Closure
5. STP O-55A-1(2) Containment Closure Verification
6. UFSAR 7.5.8
7. Technical Specifications Table 3.3.10-1
8. OI-41A Hydrogen Recombiners
9. 1C10-ALM ESFAS 14 Alarm Manual, J-09
10. ERPIP-803 Core Damage Assessment Using Hydrogen
11. EOP-8 Functional Recovery Procedure
12. EOP-13.02 Hydrogen Concentration 4.0%
13. UFSAR 1.2.5
14. UFSAR 5.1.1
15. Operating License Amendment No. 242/DPR-53 & 216/DPR-69
16. NEI 99-01 CG1

Figure C-2: Response of the TMI-2 Source Range Measurement During the First Six Hours of the Accident (ref. 10, 11)



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – RCS Temperature

Initiating Condition: Unplanned loss of decay heat removal capability

EAL:

CU4.1	**Unusual Event**
UNPLANNED event results in RCS temperature > 200°F	

Mode Applicability:

5 - Cold Shutdown, 6 - Refuel

Basis:

Generic

This EAL is a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered.

During refueling the level in the RCS will normally be maintained above the Reactor Vessel flange. Refueling evolutions that decrease water level below the Reactor Vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RCS temperatures depending on the time since shutdown.

Normal means of core temperature indication and RCS level indication may not be available in the refueling mode. Redundant means of RCS level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. Escalation to Alert would be via EAL CA3.1 based on an inventory loss or EAL CA4.1 based on exceeding its temperature duration or pressure criteria.

Plant-Specific

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F). These include TI-112C and TI-122C for forced circulation, CETs for natural circulation and TR-351 for SDC flow.

Definitions:

Unplanned

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

CCNPP Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. OP-7 Shutdown Operations
3. OP-1 Plant Startup from Cold Shutdown
4. NEI 99-01 CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – RCS Temperature

Initiating Condition: Unplanned loss of decay heat removal capability

EAL:

CU4.2	**Unusual Event**
Loss of **all** RCS temperature and RCS level indication for ≥ 15 min. (Note 4)	

Note 4: The ED 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

Mode Applicability:

5 - Cold Shutdown, 6 - Refuel

Basis:

Generic

This EAL is a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered.

During refueling the level in the RCS will normally be maintained above the Reactor Vessel flange. Refueling evolutions that decrease water level below the Reactor Vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RCS temperatures depending on the time since shutdown.

Normal means of core temperature indication and RCS level indication may not be available in the refueling mode. Redundant means of RCS level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the cold shutdown of refueling modes, this EAL would result in declaration of a UE if both temperature and level indication cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert would be via EAL CA3.1 based on an inventory loss or EAL CA4.1 based on exceeding its temperature criteria.

Plant-Specific

Reactor Vessel water level may be monitored using any of the following instruments:

- Refueling Pool Level LE-4140
- RCS Level Narrow Range LE-4138
- RCS Level Wide Range LE-4139
- Local refueling level indicator (LG-4139 and tygon tubing)
- Reactor Vessel Level Monitoring System (RVLMS)

If RCS water level will be below the bottom of the Pressurizer (48.5 ft el.), IM installs and calibrates the Refueling Level Cart in the Control Room and places the RCS Wide Range Level Monitoring System in service (LE-4139). The Wide Range Level High/Low alarms and Narrow Range Level High/Low alarms are set above/below the target RCS level. Table 1 of OP-7, Shutdown Operations, provides a cross-reference of Refueling Pool level and RVLMS alarm lights to various plant component elevations. If all RCS water level indication is lost during RCS drain down, OP-7 requires the following:

- Stop the RCS drain down and evaluate the reason for the loss of level indication.
- When at least two remote and one local RCS level indicator have been restored and are reading within 0.5 ft of each other, continue RCS drain down. Allow 15 min. for pressures to equalize. If level readings are not within 0.5 ft, call IM to investigate discrepancies.

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F). These include TI-112C and TI-122C for forced circulation, CETs for natural circulation and TR-351 for SDC flow.

CCNPP Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. OP-7 Shutdown Operations
3. OI-1I Post Accident Monitoring System Instrumentation
4. OP-1 Plant Startup from Cold Shutdown
5. NEI 99-01 CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA4.1 Alert

An UNPLANNED event results in **EITHER**:

RCS temperature > 200°F for > Table C-4 duration

OR

RCS pressure increase > 10 psi due to an UNPLANNED loss of decay heat removal capability (this condition is **not** applicable in solid plant conditions)

Table C-4 RCS Reheat Duration Thresholds		
RCS Status	CONTAINMENT CLOSURE Status	Duration
Intact **AND not** reduced inventory	N/A	60 min.*
Not intact **OR** reduced inventory	Established	20 min.*
	Not established	0 min.

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

Mode Applicability:

5 - Cold Shutdown, 6 – Refuel

Basis:

Generic

The RCS Reheat Duration Thresholds table addresses complete loss of functions required for core cooling for greater than 60 minutes during refuel and cold shutdown modes when RCS integrity is established. RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). The status of CONTAINMENT CLOSURE in this condition is immaterial given that the RCS is providing a high pressure barrier to fission product release to the environment. The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety.

The RCS Reheat Duration Thresholds table also addresses the complete loss of functions required for core cooling for greater than 20 minutes during refuel and cold shutdown modes when

Containment Closure is established but RCS integrity is not established or RCS inventory is reduced (e.g., mid-loop operation in PWRs)]. [As discussed above, RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams) . The allowed 20 minute time frame was included to allow operator action to restore the heat removal function, if possible. The allowed time frame is consistent with the guidance provided by Generic Letter 88-17, "Loss of Decay Heat Removal" (discussed later in this basis) and is believed to be conservative given that a low pressure Containment barrier to fission product release is established.

Finally, complete loss of functions required for core cooling during refuel and cold shutdown modes when neither Containment Closure nor RCS integrity are established is addressed. RCS integrity is in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). No delay time is allowed because the evaporated reactor coolant that may be released into the Containment during this heatup condition could also be directly released to the environment.

The note (*) indicates that this EAL is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the specified time frame.

The 10 psi pressure increase addresses situations where, due to high decay heat loads, the time provided to restore temperature control, should be less than 60 minutes. The RCS pressure setpoint was chosen because it is the lowest pressure that the site can read on installed Control Board instrumentation that is equal to or greater than 10 psi.

Escalation to Site Area Emergency would be via EAL CS3.1 should boiling result in significant RPV level loss leading to core uncovery.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary unplanned excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available.

The Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

Plant-Specific

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F). These include TI-112C and TI-122C for forced circulation, CETs for natural circulation and TR-351 for SDC flow.

Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components, as listed in STP O-55A, provide a functional barrier to fission product release. Containment closure is initiated by the Shift Manager if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal.

Containment closure requires, upon a loss of decay heat removal, any open penetration must be closed or capable of being closed prior to RCS boiling.

"Reduced inventory" is the condition when the reactor vessel contains irradiated fuel assemblies and RCS water level is at or below 41 ft el.

The pressure rise of greater than 10 psi infers an RCS temperature in excess of the Technical Specification cold shutdown limit (200°F) for which this EAL would otherwise permit up to sixty minutes to restore RCS cooling before declaration of an Alert (RCS intact). This EAL therefore covers situations in which it is determined that, due to high decay heat loads, the time provided to reestablish temperature control should be less than sixty minutes (as indicated by significant RCS re-pressurization).

Pressure indicators PI-103, PI-103-1 and PI-105 are capable of measuring pressure changes of 10 psi. Escalation to a Site Area Emergency would be under EAL CS3.1 should boiling result in significant Reactor Vessel level loss leading to core uncovery.

Definitions:

Containment Closure

The site specific procedurally defined actions taken to secure Containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. As applied to CCNPP, Containment Closure is the action or condition that ensures Containment and its associated systems, structures or components (SSC), as listed in STP O-55A, provide a functional barrier to fission product release.

Unplanned

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

CCNPP Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. NO-1-114 Containment Closure
3. STP O-55A-1(2) Containment Closure Verification
4. OP-7 Shutdown Operations
5. OP-1 Plant Startup from Cold Shutdown
6. NEI 99-01 CA4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Communications

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

EAL:

CU5.1 Unusual Event

Loss of **all** Table C-5 onsite (internal) communication methods affecting the ability to perform routine operations

OR

Loss of **all** Table C-5 offsite (external) communication methods affecting the ability to perform offsite notifications to any agency

Table C-5 Communications Systems		
System	**Onsite** (internal)	**Offsite** (external)
Commercial phone system	X	X
Plant page system	X	
FTS 2001 telephone system		X
CCNPP Radio System	X	
Satellite Phone System		X
Cellular Phone System	X	X

Mode Applicability:

5 - Cold Shutdown, 6 - Refuel, D - Defueled

Basis:

Generic

The purpose of this EAL is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate issues with any off-site authorities. The loss of off-site communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

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

Plant-Specific

Onsite/offsite communications systems are listed in Table C-2 (ref. 1, 2, 3).

This EAL is the cold condition equivalent of the hot condition EAL SU6.1.

CCNPP Basis Reference(s):

1. UFSAR Section 7.8
2. Emergency Response Facility Directory & Communications Equipment Information
3. NO-1-113, Control of Radio Transmitter (PRT)
4. NEI 99-01 CU6

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Inadvertent Criticality

Initiating Condition: Inadvertent criticality

EAL:

CU6.1	**Unusual Event**
An UNPLANNED sustained positive startup rate observed on nuclear instrumentation	

Mode Applicability:

5 - Cold Shutdown, 6 - Refuel

Basis:

Generic

This EAL addresses criticality events that occur in Cold Shutdown or Refueling modes such as fuel mis-loading events and inadvertent dilution events. This EAL indicates a potential degradation of the level of safety of the plant, warranting a UE classification.

Escalation would be by Emergency Director judgment.

Plant-Specific

The term "sustained" is used to allow exclusion of expected short-term positive startup rates from planned fuel bundle or control rod movements during core alteration. These short-term positive startup rates are the result of the rise in neutron population due to subcritical multiplication.

Definitions:

Unplanned

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

CCNPP Basis Reference(s):

1. UFSAR Section 7.5.2.2.
2. Technical Specifications 3.9.2
3. 1C05-ALM Reactivity Control Alarm Manual, Window D-05, D-15
4. AOP-1A Inadvertent Boron Dilution
5. NEI 99-01 CU8

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.

The events of this category pertain to the following subcategories:

1. Natural or Destructive Phenomena

Natural events include hurricanes, earthquakes or tornados that have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety. Non-naturally occurring events that can cause damage to plant facilities and include aircraft crashes, missile impacts, etc.

2. Fire or Explosion

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

3. Hazardous Gas

Non-naturally occurring events that can cause damage to plant facilities and include toxic, asphyxiant, corrosive or flammable gas leaks.

4. Security

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

5. 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.

6. Judgment

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

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU1.1 Unusual Event

Seismic event identified by **ANY two** of the following:

- Seismic Monitor indicates a seismic event detected
- Earthquake felt in plant
- National Earthquake Information Center (Note 7)

Note 7: The NEIC can be contacted by calling **(303) 273-8500**. Select **option #1** and inform the analyst you wish to confirm recent seismic activity in the vicinity of Calvert Cliffs Nuclear Power Plant. Provide the analyst with the following CCNPP coordinates: **38° 25' 39.7" north latitude, 76° 26' 45" west longitude**.

Mode Applicability:

All

Basis:

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate.

As defined in the EPRI-sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989, a "felt earthquake" is: An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of Control Room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated.

The National Earthquake Center can confirm if an earthquake has occurred in the area of the plant.

Plant-Specific

CCNPP seismic instrumentation actuates at 0.01g upon sensing any ground motion. Damage to some portions of the site may occur as a result of the felt earthquake but it should not affect the ability of safety functions to operate. This event escalates to an Alert under EAL HA1.1 if the earthquake exceeds the Operating Basis Earthquake (OBE) magnitude of 0.08 g horizontal or 0.053g vertical. (ref. 1)

The National Earthquake Information Center (NEIC) can confirm seismic activity in the vicinity of the CCNPP. The NEIC can be contacted by calling **(303) 273-8500**. Select **option #1** and inform the analyst you wish to confirm recent seismic activity in the vicinity of Calvert Cliffs Nuclear Power Plant. Provide the analyst with the following CCNPP coordinates: **38° 25' 39.7" north latitude, 76° 26' 45" west longitude** (ref. 3). Alternatively, information regarding the extent of a near-site earthquake can be obtained by calling the University of Delaware **(302) 821-1576**.

CCNPP Basis Reference(s):

1. OI-46 Seismic Measurement Equipment
2. UFSAR Section 7.5.7 Seismic Instrumentation
3. Calvert Cliffs ISFSI USAR Section 2.1.1 Site Location
4. STPI M-260-0 Seismic Instrumentation Channel Check
5. NEI 99-01 HU1
6. ECP-13-000653, Replace the existing five channel SMA-3 Seismic Montitor with a Kinemetrics Condor Seismic Monitoring System

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU1.2	**Unusual Event**
Tornado striking within PROTECTED AREA BOUNDARY	
OR	
Sustained high winds > 45 m/sec (100 mph)	

Mode Applicability:

All

Basis:

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL is based on a tornado striking (touching down) or high winds within the Protected Area.

Escalation of this emergency classification level, if appropriate, would be based on visible damage, or by other in plant conditions, via EAL HA1.2.

Plant-Specific

All Class 1 safe shutdown structures are designed for a wind velocity of 100 mph, 30 feet above ground using a gust factor of 1.1. The meteorological tower 15-minute average wind speed readings are the “sustained” wind speeds used to assess this EAL.

Definitions:

Protected Area

The site specific area which normally encompasses all controlled areas within the security Protected Area fence.

CCNPP Basis Reference(s):

1. ES-005 Civil and Structural Design Criteria
2. CCIPEEE RAN 97-031 High Winds, Floods and Other External Events Analysis
3. NEI 99-01 HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU1.3 Unusual Event

Internal flooding that has the potential to affect **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT required by Technical Specifications for the current operating mode in **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas
• Control Room • Containment • Auxiliary Building • Diesel Generator Rooms • Intake Structure • 1A/0C DG Buildings • RWT • RWT Rooms • CST No. 12 • FOST No. 21 • Auxiliary Feed Pump Rooms

Mode Applicability:

All

Basis:

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps.

Escalation of this emergency classification level, if appropriate, would be based visible damage via EAL HA1.3, or by other plant conditions.

Plant-Specific

This threshold addresses the affect of flooding caused by internal events such as component failures, Circulating, Saltwater, Component Cooling or Service Water line ruptures, equipment misalignment, fire suppression system actuation, and outage activity mishaps. The internal flooding areas contain systems that are:

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

Table H-1 Safe Shutdown Areas include all SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT needed for safe shutdown.

Safety-Related Structures, Systems and Components (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.

CCNPP Basis Reference(s):

1. CCPRA RAN: 96-024FLOOD Flood Rule Development
2. CCPRA RAN: 98-062, Internal Flood Initiating Event Frequencies
3. CCPRA RAN: 98-065, Flood Evaluations (Flood Queries)
4. 1C10-ALM ESFAS 13 Alarm Manual, J-17, CC PP RM LVL HI
5. 1C10-ALM ESFAS 13 Alarm Manual, J-18, SRW PP RM LVL HI
6. 1C10-ALM ESFAS 13 Alarm Manual, J-22, CNDSR PIT LVL HI
7. 1C10-ALM ESFAS 13 Alarm Manual, J-23, INTAKE SUMP STRUCTURE LVL HI
8. 1C10-ALM ESFAS 13 Alarm Manual, J-24, INTAKE STRUCTURE CH TRIP
9. Drawing 61502 Plant Property and Buildings
10. UFSAR Section 5A.2 Classes of Structures, Systems, and Equipment
11. NEI 99-01 HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU1.4	**Unusual Event**
Turbine failure resulting in casing penetration or damage to turbine or generator seals	

Mode Applicability:

All

Basis:

Generic

These EALs are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant.

Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIRES and flammable gas build up are appropriately classified via EAL HU2.1 and EAL HU3.1.

This EAL is consistent with the definition of a UE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.

Escalation of this emergency classification level, if appropriate, would be to EAL HA1.4 based on damage done by PROJECTILES generated by the failure or in conjunction with a steam generator tube rupture. These latter events would be classified by the Category R EALs or Category F EALs.

Plant-Specific

The turbine generator stores large amounts of rotational kinetic energy in its rotor. In the unlikely event of a major mechanical failure, this energy may be transformed into both rotational and translational energy of rotor fragments. These fragments may impact the surrounding stationary parts. If the energy-absorbing capability of these stationary turbine generator parts is insufficient, external projectiles will be released. These ejected projectiles may impact various plant structures, including those housing safety related equipment.

In the event of projectile ejection, the probability of a strike on a plant region is a function of the energy and direction of an ejected projectile and of the orientation of the turbine with respect to the plant region.

Failure of turbine or generator seals may be indicated by a loss of seal oil pressure or loss of condenser vacuum (ref. 2, 3).

CCNPP Basis Reference(s):

1. AOP-7E Main Turbine Malfunction
2. AOP-7G Loss of Condenser Vacuum
3. 1C02-ALM B-10 Seal Oil Diff Press Lo
4. NEI 99-01 HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the Protected Area

EAL:

HU1.5	**Unusual Event**
Bay water level ≥ bottom of the traveling screen cover housing (+ 120 in. Mean Sea Level) **OR** Bay water level < 13.6 ft below intake concrete level (- 43.2 in. Mean Sea Level)	

Mode Applicability:

All

Basis:

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses other site specific phenomena that can also be precursors of more serious events.

Plant-Specific

This threshold addresses high and low bay water level conditions that could be a precursor of more serious events.

Since the Intake Structure houses the saltwater pumps that are essential for safe shutdown of CCNPP, the structure was designed as a Category I structure for seismic, tornado, and hurricane conditions. The Intake Structure is also designed to protect the saltwater pump motors from external flooding due to the maximum hypothetical hurricane tide and storm surges, including wave action. The Intake Structure design loads and conditions are shown in UFSAR Table 5-7.

10 ft (+120 in.) Mean Sea Level (approximately bottom of the travelling screen cover) is the still water level used for the Intake Structural Analysis. This value was selected to be anticipatory to the design level of 18 ft Mean Sea Level (top of the travelling screen cover).

The predicted extreme low tide elevation is -3.6 ft (-43.2 in.) Mean Sea Level. However, the plant has been designed for -4.0 ft Mean Sea Level and can continue to operate with an extreme low water Elevation of -6.0 ft Mean Sea Level. The top of the saltwater pump intakes is at -9.5 ft Mean Sea Level.

Operations can measure water level from the intake concrete walking level to the Bay surface with a tape measure. This level is measured upstream (i.e., before) the trash racks. This EAL criterion is met if the water is 13.6 ft below the intake concrete level by observation. This measurement requires judgment because the Bay surface is not normally still.

Radar probes (1-LIT-1100 & 2-LIT-2100) have been installed which provide local indication (1-LI-1100 & 2-LI-2100) of Intake water level in inches relative to Mean Sea Level (ref. 4).

CCNPP Basis Reference(s):

1. UFSAR Sections 2.8.3.6 and 2.8.3.7
2. UFSAR Table 5-7
3. 1C10-ALM ESFAS 14 Alarm Manual, J-23
4. ECP-10-000208
5. NEI 99-01 HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting Vital Areas

EAL:

HA1.1 Alert

EITHER:

Seismic Monitor indicates a seismic event > OBE (0.08 g horizontal or 0.053g vertical)

OR

Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT

AND

Earthquake confirmed by **EITHER**:

Earthquake felt in plant

OR

National Earthquake Information Center (Note 7)

Note 7: The NEIC can be contacted by calling **(303) 273-8500**. Select **option #1** and inform the analyst you wish to confirm recent seismic activity in the vicinity of Calvert Cliffs Nuclear Power Plant. Provide the analyst with the following CCNPP coordinates: **38° 25' 39.7" north latitude, 76° 26' 45" west longitude**.

Mode Applicability:

All

Basis:

Generic

These EALs escalate from HU1.1 in that the occurrence of the event has resulted in damage to the safety systems in those structures evidenced by Control Room indications of degraded system response or performance. The occurrence of degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

Seismic events of this magnitude can result in a VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

The National Earthquake Information Center can confirm if an earthquake has occurred in the area of the plant.

Plant-Specific

This EAL is based on the UFSAR design Operating Basis Earthquake (OBE) of 0.08 g horizontal or 0.053 g vertical acceleration. Seismic events of this magnitude can cause damage to plant safety functions.

The method of determining whether the OBE has been exceeded relies on either the actuation of or evaluation of data from the CCNPP seismic monitor or indication of actual degraded system performance with seismic activity confirmed by shift operators on duty in the Control Room determining that the ground motion was felt or corroborated by the NEIC. According to OI-46, confirmation by one or more Control Room operators with respect to ground motion helps avoid unnecessary classification if the seismic switches inadvertently trip or detect vibrations not related to an earthquake.

CCNPP seismic instrumentation actuates at 0.01g upon sensing any seismic activity.

The National Earthquake Information Center (NEIC) can confirm seismic activity in the vicinity of the CCNPP. The NEIC can be contacted by calling **(303) 273-8500**. Select **option #1** and inform the analyst you wish to confirm recent seismic activity in the vicinity of Calvert Cliffs Nuclear Power Plant. Provide the analyst with the following CCNPP coordinates: **38° 25' 39.7" north latitude, 76° 26' 45" west longitude** (ref. 3). Alternatively, information regarding the extent of a near-site earthquake can be obtained by calling the University of Delaware **(302) 821-1576**.

Definitions:

Safety-Related Structures, Systems and Components (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.

CCNPP Basis Reference(s):

1. OI-46 Seismic Measurement Equipment
2. UFSAR Section 7.5.7 Seismic Instrumentation
3. Calvert Cliffs ISFSI USAR Section 2.1.1 Site Location
4. STPI M-260-0 Seismic Instrumentation Channel Check
5. NEI 99-01 HA1
6. ECP-13-000653, Replace the existing five channel SMA-3 Seismic Montitor with a Kinemetrics Condor Seismic Monitoring System

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting Vital Areas

EAL:

HA1.2 Alert

Tornado striking or sustained high winds > 45 m/sec (100 mph) resulting in **EITHER**:

VISIBLE DAMAGE to **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area

OR

Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas
• Control Room • Containment • Auxiliary Building • Diesel Generator Rooms • Intake Structure • 1A/0C DG Buildings • RWT • RWT Rooms • CST No. 12 • FOST No. 21 • Auxiliary Feed Pump Rooms

Mode Applicability:

All

Basis:

Generic

This EAL escalates from HU1.2 in that the occurrence of the event has resulted in visible damage to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by Control Room indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL is based on a tornado striking (touching down) or high winds that have caused visible damage to structures containing functions or systems required for safe shutdown of the plant.

Plant-Specific

This threshold addresses events that may have resulted in a Safe Shutdown Area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. Safe Shutdown Areas are vital areas that house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in cold shutdown. Safe Shutdown Areas include structures that contain the equipment of concern. The Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Safe Shutdown Areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold.

This EAL is based on the structural design basis of 100 mph. Wind loads of this magnitude can cause damage to safety functions. The meteorological tower 15-minute average wind speed readings are the "sustained" wind speeds used to assess this EAL.

Table H-1 Safe Shutdown Areas include all SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT needed for safe shutdown.

Definitions:

Safety-Related Structures, Systems and Components (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 equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

CCNPP Basis Reference(s):

1. ES-005 Civil and Structural Design Criteria
2. CCIPEEE RAN 97-031 High Winds, Floods and Other External Events Analysis Section 5.3.1
3. Drawing 61502 Plant Property and Buildings
4. UFSAR Section 5A.2 Classes of Structures, Systems, and Equipment
5. NEI 99-01 HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting Vital Areas

EAL:

HA1.3 Alert

Internal flooding in **ANY** Table H-1 area resulting in **EITHER**:

An electrical shock hazard that precludes access to operate or monitor **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area

OR

Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas
• Control Room • Containment • Auxiliary Building • Diesel Generator Rooms • Intake Structure • 1A/0C DG Buildings • RWT • RWT Rooms • CST No. 12 • FOST No. 21 • Auxiliary Feed Pump Rooms

Mode Applicability:

All

Basis:

<u>Generic</u>

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant.

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

Plant-Specific

This threshold addresses the effect of flooding caused by internal events such as component failures such as Circulating, Saltwater, Component Cooling or Service Water line ruptures, equipment misalignment, fire suppression system actuation, steam leaks or outage activity mishaps. The Internal Flooding Areas contain systems that are:

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

Uncontrolled internal flooding that has degraded safety-related equipment or created a safety hazard precluding access necessary for the safe operation or monitoring of safety equipment warrants declaration of an Alert.

Definitions:

Safety-Related Structures, Systems and Components (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.

CCNPP Basis Reference(s):

1. CCPRA RAN: 96-024FLOOD Flood Rule Development
2. CCPRA RAN: 98-062, Internal Flood Initiating Event Frequencies
3. CCPRA RAN: 98-065, Flood Evaluations (Flood Queries)
4. 1C10-ALM ESFAS 13 Alarm Manual, J-17, CC PP RM LVL HI
5. 1C10-ALM ESFAS 13 Alarm Manual, J-18, SRW PP RM LVL HI
6. 1C10-ALM ESFAS 13 Alarm Manual, J-22, CNDSR PIT LVL HI
7. 1C10-ALM ESFAS 13 Alarm Manual, J-23, INTAKE SUMP STRUCTURE LVL HI
8. 1C10-ALM ESFAS 13 Alarm Manual, J-24, INTAKE STRUCTURE CH TRIP
9. NEI 99-01 HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting Vital Areas

EAL:

HA1.4 **Alert**
Turbine failure-generated PROJECTILES resulting in **EITHER**:
VISIBLE DAMAGE to or penetration of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area
OR
Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas
• Control Room • Containment • Auxiliary Building • Diesel Generator Rooms • Intake Structure • 1A/0C DG Buildings • RWT • RWT Rooms • CST No. 12 • FOST No. 21 • Auxiliary Feed Pump Rooms

Mode Applicability:

All

Basis:

Generic

This EAL escalates from HU1.4 in that the occurrence of the event has resulted in visible damage to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by Control Room indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL addresses the threat to safety related equipment imposed by projectiles generated by main turbine rotating component failures. Therefore, this EAL is consistent with the definition of an Alert in that the potential exists for actual or substantial potential degradation of the level of safety of the plant.

Plant-Specific

The turbine generator stores large amounts of rotational kinetic energy in its rotor. In the unlikely event of a major mechanical failure, this energy may be transformed into both rotational and translational energy of rotor fragments. These fragments may impact the surrounding stationary parts. If the energy-absorbing capability of these stationary turbine generator parts is insufficient, external projectiles will be released. These ejected projectiles may impact various plant structures, including those housing safety related equipment.

Table H-1 Safe Shutdown Areas include all SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT needed for safe shutdown.

Definitions:

Projectile

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

Safety-Related Structures, Systems and Components (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 equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

CCNPP Basis Reference(s):

1. Drawing 61502 Plant Property and Buildings
2. UFSAR Section 5A.2 Classes of Structures, Systems, and Equipment
3. CCIPEEE RAN 97-031 High Winds, Floods and Other External Events Analysis Section 5.3.1
4. NEI 99-01 HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting Vital Areas

EAL:

HA1.5 Alert

Bay water level ≥ top of the traveling screen cover housing

OR

Bay water level or inside travelling screen water level < 16.0 ft below intake concrete level (-72.0 in. Mean Sea Level)

Mode Applicability:

All

Basis:

Generic

This EAL addresses other site specific phenomena that result in visible damage to vital areas or results in indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant that can also be precursors of more serious events.

Plant-Specific

This threshold covers high and low water level conditions that may have resulted in a plant Vital Area being subjected to levels beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

Since the Intake Structure houses the saltwater pumps that are essential for safe shutdown of CCNPP, the structure was designed as a Category I structure for seismic, tornado, and hurricane conditions. The Intake Structure is also designed to protect the saltwater pump motors from external flooding due to the maximum hypothetical hurricane tide and storm surges, including wave action. The Intake Structure design loads and conditions are shown in UFSAR Table 5-7.

18 ft Mean Sea Level (+216 in., top of the travelling screen cover) is the design flood level.

The predicted extreme low tide elevation is -3.6 ft Mean Sea Level. However, the plant has been designed for -4.0 ft Mean Sea Level and can continue to operate with an extreme low water Elevation of -6.0 ft (-72.0 in.) Mean Sea Level. This EAL criterion is met if the water is 16 ft below the intake concrete level by observation. This measurement requires judgment because the Bay surface is not normally still. Bay water level is upstream (i.e., before) the travelling screens. Inside travelling screen water level is considered at the Alert classification because differential pressure across the screens may depress the level below the bay water level and reduce NPSH available to the saltwater pumps. The top of the saltwater pump intake is at -9.5 ft Mean Sea Level.

Radar probes (1-LIT-1100 & 2-LIT-2100) have been installed which provide local indication (1-LI-1100 & 2-LI-2100) of Intake water level in inches relative to Mean Sea Level (ref. 4).

CCNPP Basis Reference(s):

1. UFSAR Sections 2.8.3.6 and 2.8.3.7
2. UFSAR Table 5-7
3. 1C10-ALM ESFAS 14 Alarm Manual, J-23
4. ECP-10-000208
5. NEI 99-01 HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting Vital Areas

EAL:

HA1.6 Alert

Vehicle crash resulting in **EITHER**:

VISIBLE DAMAGE to **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area

OR

Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas
• Control Room • Containment • Auxiliary Building • Diesel Generator Rooms • Intake Structure • 1A/0C DG Buildings • RWT • RWT Rooms • CST No. 12 • FOST No. 21 • Auxiliary Feed Pump Rooms

Mode Applicability:

All

Basis:

Generic

The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL addresses vehicle crashes within the Protected Area that results in visible damage to vital areas or indication of damage to safety structures, systems, or components containing functions and systems required for safe shutdown of the plant.

Plant-Specific

This EAL is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant. Vehicle types include automobiles, aircraft, trucks, cranes, forklifts, waterborne craft, etc.

Table H-1 Safe Shutdown Areas include all SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT needed for safe shutdown.

Definitions:

Safety-Related Structures, Systems and Components (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 equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

CCNPP Basis Reference(s):

1. Drawing 61502 Plant Property and Buildings
2. UFSAR Section 5A.2 Classes of Structures, Systems, and Equipment
3. NEI 99-01 HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Fire or Explosion

Initiating Condition: Fire within the Protected Area **not** extinguished within 15 min. of detection or explosion within the Protected Area

EAL:

HU2.1 Unusual Event

FIRE **not** extinguished within 15 min. of Control Room notification or verification of a Control Room fire alarm in the North Service Building, Turbine Building, Butler Building (Note 9) or **ANY** Table H-1 area (Note 4)

* Butler Building is only considered adjacent in Modes 5, 6 or D.

Note 4: The ED 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.

Note 9: Butler Building is only considered adjacent in Modes 5, 6 and D.

Table H-1 Safe Shutdown Areas
• Control Room
• Containment
• Auxiliary Building
• Diesel Generator Rooms
• Intake Structure
• 1A/0C DG Buildings
• RWT
• RWT Rooms
• CST No. 12
• FOST No. 21
• Auxiliary Feed Pump Rooms
• Service Water Rooms
• Switchgear Rooms

Mode Applicability:

All

Basis:

Generic

This EAL addresses the magnitude and extent of fires that may be potentially significant precursors of damage to safety systems. It addresses the FIRE, and not the degradation in performance of affected systems that may result.

As used here, detection is visual observation and report by plant personnel or sensor alarm indication.

The purpose of this threshold is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. As used here, notification is visual observation and report by plant personnel or sensor alarm indication. The 15-minute period to extinguish the fire begins with a credible notification that a fire is occurring or indication of a valid fire detection system alarm. Determination of a valid fire detection system alarm includes actions that can be taken within the Control Room or at nearby Fire Panels to determine that the alarm is not spurious. These actions include the use of direct or indirect indications such as redundant alarms or instrumentation readings associated with the area to ensure the alarm is not spurious and is an indication of a fire. An alarm verified in this manner is assumed to be an indication of a fire unless personnel dispatched to the scene disprove the alarm within the 15-minute period. The report, however, shall not be required to verify the alarm.

The intent of this 15 minute duration is to size the fire and to discriminate against small fires that are readily extinguished (e.g., smoldering waste paper basket).

Plant-Specific

Table H-1 Safe Shutdown Areas include all SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT needed for safe shutdown. The North Service Building, Turbine Building and Butler Building (only when in Modes 5, 6 or D) are adjacent structures.

Definitions:

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.

CCNPP Basis Reference(s):

1. Drawing 61502 Plant Property and Buildings
2. UFSAR Section 5A.2 Classes of Structures, Systems, and Equipment
3. AOP-9 Series Fire Procedures
4. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Fire or Explosion

Initiating Condition: Fire within the Protected Area **not** extinguished within 15 min. of detection or explosion within the Protected Area

EAL:

HU2.2	**Unusual Event**
EXPLOSION within the PROTECTED AREA	

Mode Applicability:

All

Basis:

Generic

This EAL addresses the magnitude and extent of explosions that may be potentially significant precursors of damage to safety systems. It addresses the explosion, and not the degradation in performance of affected systems that may result.

This EAL addresses only those explosions of sufficient force to damage permanent structures or equipment within the Protected Area.

No attempt is made to assess the actual magnitude of the damage. The occurrence of the explosion is sufficient for declaration.

The Emergency director also needs to consider any security aspects of the explosion, if applicable.

Escalation of this emergency classification level, if appropriate, would be based on EAL HA2.1.

Plant-Specific

While some explosions may also result in fires that exceed EAL HU2.1, no fire is necessary to declare an emergency in the event of an explosion. If a fire also occurs as a result or with an explosion, declare the Unusual Event based on the explosion and monitor the progress of the fire for potential escalation due to fire damage.

When used in the context of an explosion, "catastrophic failure" of a component (e.g., tank, heat exchanger, etc.) signifies a rupture of sufficient magnitude that adjacent or nearby components are affected.

Operating experience

6/25/09 Davis-Besse - A transitory Alert condition was determined to have existed based on 'Onsite Explosion Affecting Plant Operation'. A catastrophic failure-explosion of the Constant Current Potential Device (CCPD) on 'J' Bus near Air Circuit Breaker (ACB) 34563 resulted in a loss of switchyard 345 KV Bus 'J'. This event de-energized Startup Transformer 01 which is a tie from offsite sources to the Unit 13.8 KV Busses. The licensee stated that initially, the severity of the CCPD failure was not recognized because of the night time conditions and minimal lighting in the area. After daylight examination of the location of the event, it was determined that the failure of the CCPD should have been classified as an explosion affecting plant operation. Consequently, the licensee made the after-the-fact declaration.

10/27/08 Quad Cities - Quad Cities Station declared an Unusual Event due to an explosion in the protected area. The radwaste floor drain surge tank building had physical damage after reports of an explosion in the area. The damage was to the block structure of the radwaste floor drain surge tank building but the radwaste tank itself was not impacted. The cause of the explosion was still unknown but appeared to be related to a buildup of some kind of hydrocarbon gas.

Definitions:

Explosion

A rapid, violent, unconfined combustion, or catastrophic failure of pressurized/energized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

Protected Area

The site specific area which normally encompasses all controlled areas within the security Protected Area fence.

CCNPP Basis Reference(s):

1. Drawing 61502 Plant Property and Buildings
2. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Fire or Explosion

Initiating Condition: Fire or explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown

EAL:

HA2.1 Alert

FIRE or EXPLOSION resulting in **EITHER**:

VISIBLE DAMAGE to **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area

OR

Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas
• Control Room • Containment • Auxiliary Building • Diesel Generator Rooms • Intake Structure • 1A/0C DG Buildings • RWT • RWT Rooms • CST No. 12 • FOST No. 21 • Auxiliary Feed Pump Rooms

Mode Applicability:

All

Basis:

Generic

Visible damage is used to identify the magnitude of the fire or explosion and to discriminate against minor fires and explosions.

The reference to structures containing safety systems or components is included to discriminate against fires or explosions in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the fire or explosion was large enough to cause damage to these systems.

The use of visible damage should not be interpreted as mandating a lengthy damage assessment prior to classification. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency Director with the resources needed to perform detailed damage assessments.

The Emergency Director also needs to consider any security aspects of the explosion.

Escalation of this emergency classification level, if appropriate, will be based on EALs in Category S, Category F or Category R.

Plant-Specific

Table H-1 Safe Shutdown Areas include all SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT needed for safe shutdown.

Definitions:

Explosion

A rapid, violent, unconfined combustion, or catastrophic failure of pressurized/energized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

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.

Safety-Related Structures, Systems and Components (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 equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

CCNPP Basis Reference(s):

1. Drawing 61502 Plant Property and Buildings
2. UFSAR Section 5A.2 Classes of Structures, Systems, and Equipment
3. NEI 99-01 HA2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Hazardous Gas

Initiating Condition: Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal plant operations

EAL:

HU3.1	**Unusual Event**
Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS	

Mode Applicability:

All

Basis:

Generic

This EAL is based on the release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect normal plant operations.

The fact that SCBA may be worn does not eliminate the need to declare the event.

This EAL is not intended to require significant assessment or quantification. It assumes an uncontrolled process that has the potential to affect plant operations. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation.

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.

Escalation of this emergency classification level, if appropriate, would be based on EAL HA3.1.

Plant-Specific

Asphyxiant gases include carbon dioxide, smoke, etc.

Definitions:

Normal Plant Operations

Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from Normal Plant Operations.

CCNPP Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Hazardous Gas

Initiating Condition: Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal plant operations

EAL:

HU3.2	**Unusual Event**
Recommendation by local, county or state officials to evacuate or shelter site personnel based on offsite event	

Mode Applicability:

All

Basis:

Generic

Escalation of this emergency classification level, if appropriate, would be based on EAL HA3.1.

Plant-Specific

None

CCNPP Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Hazardous Gas

Initiating Condition: Access to a Vital Area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor

EAL:

HA3.1 Alert

Access to ANY of the following areas is prohibited due to toxic, corrosive, asphyxiant or flammable gases (Note 5):

- Control Room
- 45' West Electrical Penetration Rooms
- 69' Electrical Penetration Rooms
- ECCS Pump Rooms
- Charging Pump Rooms
- Component Cooling Rooms

Note 5: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then EAL HA3.1 should **not** be declared as it will have **no** adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

Mode Applicability:

All

Basis:

Generic

Gases in a Vital Area can affect the ability to safely operate or safely shutdown the reactor.

The fact that SCBA may be worn does not eliminate the need to declare the event.

Declaration should not be delayed for confirmation from atmospheric testing if the atmosphere poses an immediate threat to life and health or an immediate threat of severe exposure to gases. This could be based upon documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards.

If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown 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.

An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Flammable gasses, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This EAL assumes concentrations of flammable gasses which can ignite/support combustion.

Escalation of this emergency classification level, if appropriate, will be based on EALs in Category S, Category F or Category R.

Plant-Specific

Locations designated in the EAL are those areas that are required for Cold Shutdown that cannot be completed from the Control Room.

CCNPP Basis Reference(s):

1. NEI 99-01 HA3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Security

Initiating Condition: Confirmed security condition or threat which indicates a potential degradation in the level of safety of the plant

EAL:

HU4.1	**Unusual Event**
A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by Security Shift Supervisor **OR** A credible site-specific security threat notification **OR** A validated notification from NRC providing information of an aircraft threat	

Mode Applicability:

All

Basis:

Generic

Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as hostile actions are classifiable under EAL HA4.1, EAL HS4.1 and EAL HG4.1.

A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification level in accordance with the CCNPP Security and Safeguards Contingency Plan.

First Condition

Reference is made to the security shift supervisor because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Security and Safeguards Contingency Plan.

This threshold is based on the CCNPP Security and Safeguards Contingency Plan. The CCNPP Security and Safeguards Contingency Plan is based on guidance provided by NEI 03-12.

Second Condition

This threshold is included to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the Unusual Event.

The determination of "credible" is made through use of information found in the CCNPP Security and Safeguards Contingency Plan .

Third Condition

The intent of this EAL is to ensure that notifications for the aircraft threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

This EAL is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Unusual Event.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Escalation to Alert emergency classification level via EAL HA4.1 would be appropriate if the threat involves an airliner within 30 minutes of the plant.

Plant-Specific

If the Security Shift Supervisor determines that a threat notification is credible, the Security Shift Supervisor will notify the Operations Shift Manager that a "Credible Threat" condition exists for Calvert Cliffs. Generally, Calvert Cliffs Security Procedures address standard practices for determining credibility. The three main criteria for determining credibility are: technical feasibility, operational feasibility, and resolve. For Calvert Cliffs, a validated notification delivered by the FBI, NRC or similar agency is treated as credible.

Definitions:

Airliner/Large Aircraft

Any size or type of aircraft with the potential for causing significant damage to the plant (refer to the Security Plan for a more detailed definition).

Hostile Action

An act toward CCNPP 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 CCNPP. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

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.

CCNPP Basis Reference(s):

1. CCNPP Security and Safeguards Contingency Plan
2. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Security

Initiating Condition: Hostile action within the Owner Controlled Area or airborne attack threat

EAL:

HA4.1 Alert

A HOSTILE ACTION is occurring or has occurred within the Owner Controlled Area as reported by Security Shift Supervisor

OR

A validated notification from NRC of an AIRLINER attack threat within 30 min. of the site

Mode Applicability:

All

Basis:

Generic

Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering).

First Condition

This condition addresses the potential for a very rapid progression of events due to a hostile action. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the Owner Controlled Area. Those events are adequately addressed by other EALs.

Note that this condition is applicable for any hostile action occurring, or that has occurred, in the Owner Controlled Area including the Independent Spent Fuel Storage Installation (ISFSI).

Second Condition

This condition addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time.

The intent of this condition is to ensure that notifications for the airliner attack threat are made in a timely manner and that Offsite Response Organizations (OROs) and plant personnel are at a state of heightened awareness regarding the credible threat. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.

This condition is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Plant-Specific

Definitions:

Airliner/Large Aircraft

Any size or type of aircraft with the potential for causing significant damage to the plant (refer to the Security Plan for a more detailed definition).

Hostile Action

An act toward CCNPP 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 CCNPP. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

CCNPP Basis Reference(s):

1. CCNPP Security and Safeguards Contingency Plan
2. NEI 99-01 HA4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Security

Initiating Condition: Hostile action within the Protected Area

EAL:

HS4.1	**Site Area Emergency**
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by Security Shift Supervisor	

Mode Applicability:

All

Basis:

Generic

This condition represents an escalated threat to plant safety above that contained in the Alert in that a hostile force has progressed from the Owner Controlled Area to the Protected Area.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires Offsite Response Organization (ORO) readiness and preparation for the implementation of protective measures.

This EAL addresses the potential for a very rapid progression of events due to a hostile action. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the Protected Area. Those events are adequately addressed by other EALs.

Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack.

Plant-Specific

A hostile action that occurs or has occurred within the ISFSI area is not classified under this EAL. The ISFSI is located in the OCA and hostile action occurring in the ISFSI would be classified under HA4.1. If hostile action in the ISFSI results in damage to a cask confinement boundary, the event would meet the Unusual Event classification threshold for EAL EU1.1 as well as HA4.1.

Definitions:

Hostile Action

An act toward CCNPP 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 CCNPP. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

Protected Area

The site specific area which normally encompasses all controlled areas within the security Protected Area fence.

CCNPP Basis Reference(s):

1. CCNPP Security and Safeguards Contingency Plan
2. NEI 99-01 HS4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Security

Initiating Condition: Hostile action resulting in loss of physical control of the facility

EAL:

HG4.1 General Emergency

A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain **ANY** of the following safety function acceptance criteria:

- Reactivity control (RC)
- Vital Auxiliaries (VA)
- RCS pressure and inventory control (PIC)
- Core & RCS heat removal (HR)

Mode Applicability:

All

Basis:

Generic

This EAL encompasses conditions under which a hostile action has resulted in a loss of physical control of Vital Areas (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location.

If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the threshold is not met.

Plant-Specific

Safety functions of concern in this EAL include:

- Reactivity control
- RCS inventory control
- RCS pressure control
- Core & RCS heat removal

These safety functions are maintained by meeting the relevant EOP Safety Function Acceptance Criteria (ref. 1).

Definitions:

Hostile Action

An act toward CCNPP 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 CCNPP. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

CCNPP Basis Reference(s):

1. CCNPP Security and Safeguards Contingency Plan
2. CEN-152 Combustion Engineering Emergency Procedure Guidelines
3. NEI 99-01 HG1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Security

Initiating Condition: Hostile action resulting in loss of physical control of the facility

EAL:

HG4.2	**General Emergency**
A HOSTILE ACTION has caused failure of Spent Fuel Cooling systems	
AND	
IMMINENT fuel damage is likely	

Mode Applicability:

All

Basis:

Generic

This EAL addresses failure of spent fuel cooling systems as a result of hostile action if imminent fuel damage is likely.

Plant-Specific

Definitions:

Hostile Action

An act toward CCNPP 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 CCNPP. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

Imminent

Mitigation actions have been ineffective, additional actions are not expected to be successful, and trended information indicates that the event or condition will occur. Where imminent timeframes are specified, they shall apply.

CCNPP Basis Reference(s):

1. CCNPP Security and Safeguards Contingency Plan
2. NEI 99-01 HG1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Control Room Evacuation

Initiating Condition: Control Room evacuation has been initiated

EAL:

HA5.1	**Alert**
Control Room evacuation has been initiated	

Mode Applicability:

All

Basis:

Generic

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facilities may be necessary.

Inability to establish plant control from outside the Control Room will escalate this event to a Site Area Emergency.

Plant-Specific

AOP-9A Control Room Evacuation and Safe Shutdown Due to a Severe Control Room Fire and AOP-11 Control Room Evacuation and Safe Shutdown - Non-Fire Conditions provide specific instructions for evacuating the Control Room/Building and establishing plant control in alternate locations.

CCNPP Basis Reference(s):

1. AOP-9A Control Room Evacuation and Safe Shutdown Due to a Severe Control Room Fire
2. AOP-11 Control Room Evacuation and Safe Shutdown - Non-Fire Conditions
3. NEI 99-01 HA5

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Control Room Evacuation

Initiating Condition: Control Room evacuation has been initiated and plant control **cannot** be established

EAL:

HS5.1	**Site Area Emergency**

Control Room evacuation has been initiated **AND EITHER**:

Inability to establish Auxiliary Feedwater to at least one steam generator within 30 min. (Note 4)

OR

Inability to establish reactor coolant make-up (charging pump flow) within 60 min. (Note 4)

Note 4: The ED 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.

Mode Applicability:

All

Basis:

Generic

The intent of this EAL is to capture those events where control of the plant cannot be reestablished in a timely manner. In this case, expeditious transfer of control of safety systems has not occurred (although fission product barrier damage may not yet be indicated).

The intent of the EAL is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to shutdown the reactor and maintain it shutdown), reactor water level (ability to cool the core), and decay heat removal (ability to maintain a heat sink).

The determination of whether or not control is established at the remote shutdown panel is based on Emergency Director (ED) judgment. The Emergency Director is expected to make a reasonable, informed judgment within the site specific time for transfer that the licensee has control of the plant from the remote shutdown panel.

Escalation of this emergency classification level, if appropriate, would be by EALs in Category F or Category R.

Plant-Specific

AOP-9A Control Room Evacuation and Safe Shutdown Due to a Severe Control Room Fire and AOP-11 Control Room Evacuation and Safe Shutdown - Non-Fire Conditions provide specific instructions for evacuating the Control Room/Building and establishing plant control in alternate locations.

An analysis was performed to determine how quickly control must be re-established at CCNPP without core uncovery or damage. A RETRAN simulation shows that the steam generators go dry at about 47 minutes for the AOP-9 (station fire) scenario. RCS pressure reaches the lowest pressurizer safety valve setpoint soon thereafter. Restoring feedwater within 45 minutes assures that RCS pressure remains below the safety valve setpoint thus avoiding inventory loss. The maximum time allowable to restore RCS inventory for Appendix R (station fire) scenarios is 90 minutes. Site Emergency declaration at 30 minutes and 60 minutes for inability to restore feedwater and RCS make-up respectively thus constitutes a conservative action for emergency response.

This EAL is based on analysis and actual procedure walk throughs. Licensee Event Report (LER) 50-371/89-009, Rev. 2, (transmitted to the NRC on July 7, 1989) documents the analysis that demonstrates the ability to safely shutdown Unit 1 in accordance with AOP-9.

CCNPP Basis Reference(s):

1. AOP-9A Control Room Evacuation and Safe Shutdown Due to a Severe Control Room Fire
2. AOP-11 Control Room Evacuation and Safe Shutdown - Non-Fire Conditions
3. Letter, L.B. Russell (BG&E) to James H. Joyner (U.S. Nuclear Regulatory Commission Region I), Emergency Action Level Review Meeting, June 6, 1991
4. NEI 99-01 HS2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Judgment

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

EAL:

HU6.1	**Unusual Event**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which: Indicate a potential degradation of the level of safety of the plant **OR** Indicate a security threat to facility protection has been initiated **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs	

Mode Applicability:

All

Basis:

Generic

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

Plant-Specific

None

CCNPP Basis Reference(s):

1. NEI 99-01 HU5

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Judgment

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

EAL:

HA6.1 Alert Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve: An actual or potential substantial degradation of the level of safety of the plant **OR** A security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION **ANY** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels (1,000 mRem TEDE and 5,000 mRem thyroid CDE)

Mode Applicability:

All

Basis:

Generic

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

Plant-Specific

Definitions:

Hostile Action

An act toward CCNPP 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 CCNPP. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

CCNPP Basis Reference(s):

1. NEI 99-01 HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Judgment

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

EAL:

HS6.1 Site Area Emergency

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

Actual or likely major failures of plant functions needed for protection of the public

OR

HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public

ANY releases are **not** expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1,000 mRem TEDE and 5,000 mRem thyroid CDE) beyond the site boundary

Mode Applicability:

All

Basis:

Generic

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

Plant-Specific

Definitions:

Hostile Action

An act toward CCNPP 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 CCNPP. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

CCNPP Basis Reference(s):

1. NEI 99-01 HS3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Judgment

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

EAL:

HG6.1	**General Emergency**

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

Actual or IMMINENT substantial core degradation or melting with potential for loss of Containment integrity

OR

HOSTILE ACTION that results in an actual loss of physical control of the facility.

Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1,000 mRem TEDE and 5,000 mRem thyroid CDE) offsite for more than the immediate site area

Mode Applicability:

All

Basis:

Generic

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

Plant-Specific

Definitions:

Hostile Action

An act toward CCNPP 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 CCNPP. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

Imminent

Mitigation actions have been ineffective, additional actions are not expected to be successful, and trended information indicates that the event or condition will occur. Where imminent timeframes are specified, they shall apply.

CCNPP Basis Reference(s):

1. NEI 99-01 HG2

Category S – System Malfunction

EAL Group: Hot Conditions (RCS temperature > 200ºF); EALs in this category are applicable only in one or more hot operating modes. Numerous system-related equipment failure events that warrant emergency classification have been identified in this category. They may pose actual or potential threats to plant safety.

The events of this category pertain to the following subcategories:

1. Loss of 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 the 4 kV safeguard buses.

2. Loss of 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 the 125 VDC buses.

3. Criticality & RPS Failure

Inadvertent criticalities pose potential personnel safety hazards as well being indicative of losses of reactivity control.

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

4. Inability to Reach or Maintain Shutdown Conditions

System malfunctions may lead to failure of the plant to be brought to the required plant operating condition required by technical specifications if a limiting condition for operation (LCO) is not met.

5. Instrumentation

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

6. Communications

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

7. Fuel Clad Degradation

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

8. RCS Leakage

The Reactor Vessel provides a volume for the coolant that covers the reactor core. The Reactor 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 are utilized to indicate potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and Containment integrity.

Category: S – System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: Loss of **all** offsite AC power to 4kV vital buses for ≥ 15 min.

EAL:

SU1.1	**Unusual Event**
Loss of **all offsite** AC power, Table S-1, to 4kV vital buses 11(21) and 14(24) for ≥ 15 min. (Note 4)	

Note 4: The ED 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.

Table S-1 AC Power Sources	
Onsite	• 1(2)A DG • 1(2)B DG • 0C DG , if aligned
Offsite	• 500kV transmission line 5051* • 500kV transmission line 5052* • 500kV transmission line 5072* • SMECO line , if aligned * A credited 500kV line must have an independent 13kV service transformer

Mode Applicability:

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

Basis:

Generic

Prolonged loss of off-site AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of AC power to emergency busses.

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

Plant-Specific

CCNPP essential buses are 4kV vital buses 11(21) and 14(24). There are five offsite power sources available to these buses:

- Three 500kV transmission lines (Lines 5051, 5052, and 5072) supply offsite power to the 500kV switchyard via the transmission network.
- One 69kV/13kV Southern Maryland Electric Cooperative (SMECO) line may be manually connected to either 13kV bus and then to the 4kV vital buses. Under certain operational conditions, 13kV bus(es) may be receiving power from SMECO or may be quickly connected to the SMECO tie-line. The SMECO line is not used to carry loads for an operating unit and may provide power to no more than two 4kV vital buses simultaneously.
- If a fault affects only one unit, power may be obtained from the 500kV supply of the unaffected unit through a single 13kV transformer. This is considered an offsite AC power source available to the affected unit.

Based on operational experience, if the SMECO line or the 0C DG is not already aligned, these cannot be considered available/capable of supplying the bus due to the time it will take to align them. In any case, if this cannot be accomplished within 15 minutes, they are not available and an Unusual Event must be declared.

In-house power is fed back from the 500kV ring bus through 2 13kV transformers (designated P-13000-1 and P-13000-2). Normally P-13000-1 supplies all of Unit -1 (except 14 4kV bus) and 21 4kV bus. P-13000-2 supplies all of Unit-2 (except 21 4kV bus) and 14 4kV bus. Either P-13000 is capable of supplying all loads on both Units.

The fifteen-minute interval was selected as a threshold to exclude transient power losses.

CCNPP Basis Reference(s):

1. UFSAR Section 8 and Figure 8-1
2. Technical Specifications LCO 3.8.1 AC Sources-Operating
3. Technical Specifications LCO 3.8.9 Distribution Systems-Operating
4. STP-O-90 AC Sources and On-site Power Distribution Systems 7 Day Operability Verification
5. EOP-2 Loss of Off-site Power
6. NEI 99-01 SU1

Category: S – System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: AC power capability to 4kV vital buses reduced to a single power source for ≥15 min. such that **ANY** additional single failure would result in a complete loss of all 4kV vital bus power

EAL:

SA1.1 Alert

AC power capability to 4kV vital buses 11(21) and 14(24) reduced to a single power source, Table S-1, for ≥ 15 min. (Note 4)

AND

ANY additional single power source failure will result in a complete loss of all 4kV vital bus power

Note 4: The ED 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.

Table S-1 AC Power Sources	
Onsite	• 1(2)A DG • 1(2)B DG • 0C DG , if aligned
Offsite	• 500kV transmission line 5051* • 500kV transmission line 5052* • 500kV transmission line 5072* • SMECO line , if aligned * A credited 500kV line must have an independent 13kV service transformer

Mode Applicability:

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

Basis:

Generic

The condition indicated by this EAL is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a complete loss of 4kV vital bus AC power to one or both units. This condition could occur due to a loss of off-site power with a concurrent failure of all but one emergency generator to supply power to its emergency busses. Another related condition could be the loss of all off-site power and loss of on-site emergency generators with only one train of 4kV vital busses being backfed from the unit main generator, or the loss of on-site emergency generators with only one train of 4kV vital busses being backfed from off-site power. The subsequent loss of this single power source would escalate the event to a Site Area Emergency in accordance with EAL SS1.1.

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

Plant-Specific

CCNPP essential buses are 4kV vital buses 11(21) and 14(24). There are five offsite power sources available to these buses:

- Three 500kV transmission lines (Lines 5051, 5052, and 5072) supply offsite power to the 500kV switchyard via the transmission network.
- One 69kV/13kV Southern Maryland Electric Cooperative (SMECO) line may be manually connected to either 13kV bus and then to the 4kV vital buses. Under certain operational conditions, 13kV bus(es) may be receiving power from SMECO or may be quickly connected to the SMECO tie-line. The SMECO line is not used to carry loads for an operating unit and may provide power to no more than two 4kV vital buses simultaneously.
- If a fault affects only one unit, power may be obtained from the 500kV supply of the unaffected unit through a single 13 kV transformer. This is considered an offsite AC power source available to the affected unit.

Based on operational experience, if the SMECO line or the 0C DG is not already aligned, these cannot be considered available/capable of supplying the bus due to the time it will take to align them. In any case, if this cannot be accomplished within 15 minutes, they are not available and the appropriate emergency classification must be declared.

In-house power is fed back from the 500kV ring bus through 2 13kV transformers (designated P-13000-1 and P-13000-2). Normally P-13000-1 supplies all of Unit -1 (except 14 4KV bus) and 21 4kV bus. P-13000-2 supplies all of Unit-2 (except 21 4kV bus) and 14 4kV bus. Either P-13000 is capable of supplying all loads on both Units.

There are five onsite AC power sources:

- 1A DG for bus 11
- 1B DG for bus 14
- 2A DG for bus 21
- 2B DG for bus 24
- 0C DG may be aligned to any vital 4kV bus on either unit.

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If the capability for multiple sources to energize the unit vital buses within 15 minutes is not restored, an Alert is declared under this EAL. The subsequent loss of the single remaining power source escalates the event to a Site Area Emergency under EAL SS1.1.

CCNPP Basis Reference(s):

1. UFSAR Section 8 and Figure 8-1
2. Technical Specifications LCO 3.8.1 AC Sources-Operating
3. Technical Specifications LCO 3.8.9 Distribution Systems-Operating
4. OI-21A-1 1A Diesel Generator
5. OI-21A-2 2A Diesel Generator
6. OI-21B-1 1B Diesel Generator
7. OI-21B-2 2B Diesel Generator
8. OI-21C OC Diesel Generator
9. STP-O-90 AC Sources and On-site Power Distribution Systems 7 Day Operability Verification
10. AOP-7I Loss of 4kV, 480 Volt, or 208/120 Volt Instrument Bus Power
11. AOP-3F Loss of Off-site Power While in MODES 3, 4, 5, or 6
12. EOP-0 Post-trip Immediate Actions
13. EOP-2 Loss of Off-site Power
14. EOP-7 Station Blackout
15. EOP-8 Functional Recovery
16. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: Loss of **all** offsite and **all** onsite AC power to 4kV vital buses for ≥ 15 min.

EAL:

SS1.1 Site Area Emergency

Loss of **all** offsite and **all** onsite AC power, Table S-1, to 4kV vital buses 11(21) and 14(24) for ≥ 15 min. (Note 4)

Note 4: The ED 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.

Table S-1 AC Power Sources	
Onsite	• 1(2)A DG • 1(2)B DG • 0C DG , if aligned
Offsite	• 500kV transmission line 5051* • 500kV transmission line 5052* • 500kV transmission line 5072* • SMECO line , if aligned * A credited 500kV line must have an independent 13kV service transformer

Mode Applicability:

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

Basis:

Generic

Loss of all AC power to emergency busses compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power to 4kV vital busses will lead to loss of Fuel Clad, RCS, and Containment, thus this event can escalate to a General Emergency.

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

Escalation to General Emergency is via EALs in Category F or EAL SG1.1.

Plant-Specific

The CCNPP vital buses are 4kV buses 11(21) and 14(24). There are five offsite power sources available to these buses:

- Three 500kV transmission lines (Lines 5051, 5052, and 5072) supply offsite power to the 500kV switchyard via the transmission network.
- One 69kV/13kV Southern Maryland Electric Cooperative (SMECO) line may be manually connected to either vital 13kV bus and then to the 4kV vital buses. Under certain operational conditions, 13kV bus(es) may be receiving power from SMECO or may be quickly connected to the SMECO tie-line. The SMECO line is not used to carry loads for an operating unit and may provide power to no more than two 4kV vital buses simultaneously.
- If a fault affects only one unit, power may be obtained from the 500kV supply of the unaffected unit through a single 13kV transformer. This is considered an offsite AC power source available to the affected unit.

In-house power is fed back from the 500kV ring bus through 2 13kV transformers (designated P-13000-1 and P-13000-2). Normally P-13000-1 supplies all of Unit -1 (except 14 4kV bus) and 21 4kV bus. P-13000-2 supplies all of Unit-2 (except 21 4kV bus) and 14 4kV bus. Either P-13000 is capable of supplying all loads on both Units.

There are five onsite AC power sources:

- 1A DG for bus 11
- 1B DG for bus 14
- 2A DG for bus 21
- 2B DG for bus 24
- 0C DG may be aligned to any vital 4kV bus on either unit.

Based on operational experience, if the SMECO line or the 0C DG is not already aligned, these cannot be considered available/capable of supplying the bus due to the time it will take to align them. In any case, if this cannot be accomplished within 15 minutes, they are not available and a Site Area Emergency must be declared.

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of all AC power to vital buses. Even though an essential bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus then the bus should not be considered operable.

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If the 0C DG is available but is not powering a vital bus within 15 minutes, the EAL remains applicable.

CCNPP Basis Reference(s):

1. UFSAR Section 8 and Figure 8-1
2. Technical Specifications LCO 3.8.1 AC Sources-Operating
3. Technical Specifications LCO 3.8.9 Distribution Systems-Operating
4. OI-21A-1 1A Diesel Generator
5. OI-21A-2 2A Diesel Generator
6. OI-21B-1 1B Diesel Generator
7. OI-21B-2 2B Diesel Generator
8. OI-21C OC Diesel Generator
9. STP-O-90 AC Sources and On-site Power Distribution Systems 7 Day Operability Verification
10. AOP-7I Loss of 4kV, 480 Volt, or 208/120 Volt Instrument Bus Power
11. AOP-3F Loss of Off-site Power While in MODES 3, 4, 5, or 6
12. EOP-0 Post-trip Immediate Actions
13. EOP-2 Loss of Off-site Power
14. EOP-7 Station Blackout
15. EOP-8 Functional Recovery
16. NEI 99-01 SS1

Category: S –System Malfunction

Subcategory: 1 – Loss of Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to 4kV vital buses

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power, Table S-1, to 4kV vital buses 11(21) and 14(24)

AND EITHER:

Restoration of at least one 4kV vital bus within 4 hours is **not** likely

OR

CET readings > 700°F

Table S-1 AC Power Sources	
Onsite	• 1(2)A DG • 1(2)B DG • 0C DG , if aligned
Offsite	• 500kV transmission line 5051* • 500kV transmission line 5052* • 500kV transmission line 5072* • SMECO line , if aligned * A credited 500kV line must have an independent 13kV service transformer

Mode Applicability:

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

Basis:

Generic

Loss of all AC power to emergency busses compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power to emergency busses will lead to loss of fuel clad, RCS, and Containment, thus warranting declaration of a General Emergency.

This EAL is specified to assure that in the unlikely event of a prolonged loss of all AC power to vital 4kV buses, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory.

The likelihood of restoring at least one vital bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

In addition, under these conditions, fission product barrier monitoring capability may be degraded.

Plant-Specific

The CCNPP vital buses are 4kV buses 11(21) and 14(24). There are five offsite power sources available to these buses:

- Three 500kV transmission lines (Lines 5051, 5052, and 5072) supply offsite power to the 500kV switchyard via the transmission network.
- One 69kV/13kV Southern Maryland Electric Cooperative (SMECO) line may be manually connected to either vital 13kV bus and then to the 4kV vital buses. Under certain operational conditions, 13kV bus(es) may be receiving power from SMECO or may be quickly connected to the SMECO tie-line. The SMECO line is not used to carry loads for an operating unit and may provide power to no more than two 4kV vital buses simultaneously.
- If a fault affects only one unit, power may be obtained from the 500kV supply of the unaffected unit through a single 13kV transformer. This is considered an offsite AC power source available to the affected unit.

In-house power is fed back from the 500kV ring bus through 2 13kV transformers (designated P-13000-1 and P-13000-2). Normally P-13000-1 supplies all of Unit -1 (except 14 4kV bus) and 21 4kV bus. P-13000-2 supplies all of Unit-2 (except 21 4kV bus) and 14 4kV bus. Either P-13000 is capable of supplying all loads on both Units.

There are five onsite AC power sources:

- 1A DG for bus 11
- 1B DG for bus 14
- 2A DG for bus 21
- 2B DG for bus 24

- 0C DG may be aligned to any vital 4kV bus on either unit.

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of all AC power to vital buses. Even though a vital bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus then the bus should not be considered operable.

CCNPP is licensed both for a four hour SBO coping category and a one hour SBO coping category. The ability of each unit to cope with a four hour SBO duration was based on an assessment of condensate inventory required for decay heat removal, Class 1E battery capacity, compressed air availability or manual operation of certain valves, effects of loss of ventilation, Containment isolation valve operability, and reactor coolant inventory loss. A plant-specific analysis indicates that the expected rates of reactor coolant inventory loss under SBO conditions do not result in core uncovery in a SBO for four hours. Therefore, makeup systems in addition to those currently available under SBO conditions are not required to maintain core cooling under natural circulation (including reflux cooling). Thus, conditions in which restoration of AC power within four hours is not likely are included in the EAL.

Installation of the SBO diesel allowed CCNPP to operate as a plant having a one hour coping capability. This allowance is in recognition that sufficient diesel generator back-up reduces the likelihood of station blackout. The analysis for the four hour coping category however, provides the source of the appropriate estimate of the time to core uncovery following SBO from which the plant cannot recover. This time (four hours) is therefore used as the basis for determining when to declare a General Emergency subsequent to a prolonged SBO.

Core Exit Thermocouples (CETs) are a component of the Inadequate Core Cooling Instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. A superheat condition is indicated by CET readings above 700ºF. The RCS Pressure Safety Limit is 2750 psia per CCNPP Technical Specifications. The saturation temperature for this pressure is 682.2°F. Per Action Value Bases Document EOP-24.33, the uncertainty on CET Temperature is +/- 39.8°F. If one or more CETs indicate 722°F (682.2 + 39.8), subcooling has been lost for at least some locations in the core. CET indications at or above 722°F are a clear sign that core heat removal capability is lost or greatly reduced and one fission product barrier, the fuel clad, is threatened due to elevated fuel temperatures. 700°F qualifies as a condition representing a potential loss of the fuel clad barrier (ref. 16).

CCNPP Basis Reference(s):

1. UFSAR Section 8 and Figure 8-1
2. Technical Specifications LCO 3.8.1 AC Sources-Operating
3. Technical Specifications LCO 3.8.9 Distribution Systems-Operating
4. OI-21A-1 1A Diesel Generator
5. OI-21A-2 2A Diesel Generator
6. OI-21B-1 1B Diesel Generator
7. OI-21B-2 2B Diesel Generator
8. OI-21C OC Diesel Generator
9. STP-O-90 AC Sources and On-site Power Distribution Systems 7 Day Operability Verification
10. AOP-7I Loss of 4kV, 480 Volt, or 208/120 Volt Instrument Bus Power
11. AOP-3F Loss of Off-site Power While in MODES 3, 4, 5, or 6
12. EOP-0 Post-trip Immediate Actions
13. EOP-2 Loss of Off-site Power
14. EOP-7 Station Blackout
15. EOP-8 Functional Recovery
16. EOP-24.33 Action Value Bases Document
17. ERPIP-800 Core Damage Assessment
18. ERPIP-802 Core Damage Assessment Using Core Exit Thermocouples
19. EOP-5 Loss of Coolant Accident
20. CEN-152 Emergency Procedure Guidelines
21. OP-7 Shutdown Operations
22. ERPIP-601 Severe Accident Management Initial Diagnosis
23. Letter dated March 6, 1997 from Charles H. Cruse to USNRC "Revision to Emergency Action Levels Technical Basis Document"
24. NEI 99-01 SG1

Category: S – System Malfunction

Subcategory: 2 – Loss of DC Power

Initiating Condition: Loss of **all** vital DC power for ≥ 15 min.

EAL:

SS2.1	**Site Area Emergency**
< 105 VDC on **all** 125 VDC buses (11, 12, 21 and 22) for ≥ 15 min. (Note 4)	

Note 4: The ED 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.

Mode Applicability:

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

Basis:

Generic

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of Containment integrity when there is significant decay heat and sensible heat in the reactor system.

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

Escalation to a General Emergency would occur by EALs in Category R and Category F.

Plant-Specific

The 125 VDC vital system is divided into four independent and isolated channels. Each channel consists of one battery, two battery chargers, one DC bus, multiple DC unit control panels, and two inverters. Each inverter has an associated vital AC distribution panel board. Power to the DC bus, DC unit control panels, and inverters is supplied by the station batteries and/or the battery chargers. Each battery charger is fully rated and can recharge a discharged battery while at the same time supplying the steady state power requirements of the system. A reserve 125 VDC system for the plant is completely independent and isolated from all four separation groups, yet is capable of replacing any of the 125 VDC batteries. This system consists of one battery, one battery charger, and the associated DC switching equipment. Only the battery may be transferred for replacement duty.

The safety-related station batteries have been sized to carry their expected shutdown loads following a plant trip/LOCA and loss of offsite power or following a station blackout without battery terminal voltage falling below 105 volts. The loss of the 1A Diesel Generator 125 VDC bus 14 or 0C Diesel Generator bus 16 does not constitute an entry condition for this EAL.

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

CCNPP Basis Reference(s):

1. UFSAR Section 8.4.3 and Figure 8.9
2. EOP-0 Post-Trip Immediate Actions
3. EOP-2 Loss of Off-Site Power, Section V
4. AOP-7J Loss of 120 Volt Vital AC or 125 Volt Vital DC Power
5. Technical Specifications Bases 3.8.4
6. NEI 99-01 SS3

Category: S – System Malfunction

Subcategory: 3 – Criticality & RPS Failure

Initiating Condition: Inadvertent criticality

EAL:

SU3.1	**Unusual Event**
An UNPLANNED sustained positive startup rate observed on nuclear instrumentation	

Mode Applicability:

3 - Hot Standby, 4 - Hot Shutdown

Basis:

Generic

This EAL addresses inadvertent criticality events. While the primary concern of this EAL is criticality This EAL addresses inadvertent criticality events. This EAL indicates a potential degradation of the level of safety of the plant, warranting a UE classification. This EAL excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated).

Escalation would be by EALs in Category F, as appropriate to the operating mode at the time of the event.

Plant-Specific

The term “sustained” is used to allow exclusion of expected short-term positive startup rates from planned fuel bundle or control rod movements during core alteration. These short-term positive startup rates are the result of the rise in neutron population due to subcritical multiplication.

Definitions:

Unplanned

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

CCNPP Basis Reference(s):

1. UFSAR Section 7.5.2
2. 1C05-ALM Reactivity Control Alarm Manual, Window D-05, D-15
3. AOP-1A Inadvertent Boron Dilution
4. NEI 99-01 SU8

Category: S – System Malfunction

Subcategory: 3 – Criticality & RPS Failure

Initiating Condition: Automatic trip failed to shut down the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor

EAL:

SA3.1 Alert
An automatic reactor trip failed to shut down the reactor
AND
Manual actions taken at the Control Room panels successfully shut down the reactor as indicated by reactor power ≤ 5%

Mode Applicability:

1 - Power Operation

Basis:

Generic

The reactor should be considered shutdown when it producing less heat than the maximum decay heat load for which the safety systems are designed (5% power).

Manual trip actions taken at the Control Room panels are any set of actions by the reactor operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

This condition indicates failure of the automatic protection system to trip the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient. Thus the plant safety has been compromised because design limits of the fuel may have been exceeded. An Alert is indicated because conditions may exist that lead to potential loss of fuel clad barrier or RCS barrier and because of the failure of the Reactor Protection System to automatically shut down the plant.

If manual actions taken at the reactor control console fail to shut down the reactor, the event would escalate to a Site Area Emergency.

Plant-Specific

Following a successful reactor trip, nuclear power promptly drops to about six percent of the original power level and then decays to a level some 8 decades less at a startup rate (SUR) of about (-)1/3 DPM. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from an automatic reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation (NI) and a negative SUR as nuclear power drops into the source range.

The RPS setpoints listed in Figure S-1 should result in an automatic reactor trip:

Figure S-1 Automatic RPS Trips

REACTOR TRIP		
High Power Level	2/4	Variable
High Rate-of-Change of Power	2/4 below 15% Pwr	2.6 decade/min.
Low Reactor Coolant Flow	2/4 above 10^{-4}% Pwr	Variable
Low Steam Generator Pressure	2/4	670 psig
Low Steam Generator Water Level	2/4	10 in. below top of feed ring
High Pressurizer Pressure	2/4	2385 psig
Thermal Margin/Low Pressure	2/4 above 10^{-4}% Pwr	Variable
Loss of Load	2/4 above 15% Pwr	N/A
High Containment Pressure	2/4	4 psig
Axial Flux Offset	2/4	Variable
Thermal Margin/SG Press. Diff. Hi	2/4 above 10^{-4}% Pwr	135 psid

Per EOP-0, Post-Trip Immediate Actions, the operator ensures that the reactor has tripped by depressing one set of Manual Reactor Trip buttons immediately following any symptoms of a reactor trip. The symptoms include:

- Reactor Trip alarm
- Control Element Assembly (CEA) Circuit Breaker(s) Trip alarms
- Rapid Lowering in Reactor Power
- Protection Channel Trip alarm
- Reactor Protective System (RPS) Trip Bistable Lights lit

Following depression of the reactor trip buttons, the operator verifies that reactor power is decreasing. If these responses cannot be verified, as part of contingency actions, the operator is instructed to open the motor generator (MG) set feeder breakers that provide power to the Control Element Drive Mechanism (CEDM).

If reactor power is above 5%, the reactor is producing more heat than the Auxiliary Feedwater system and Atmospheric Dump Valves are designed to remove (ref. 6, 7, 8). The Alert emergency classification is required whenever the Shift Manager determines that a required automatic reactor trip did not succeed in reducing reactor power to 5% or lower. It is recognized that EOP-0 instructs the operator to depress the manual trip buttons whether or not a required automatic reactor trip actually occurred. However, the failure of the RPS to complete a reactor trip that reduced reactor power to 5% or lower following receipt of an automatic trip signal meets the Alert classification threshold of potential substantial degradation in the level of safety of the plant. This is true even if no radiation alarms indicate fuel problems.

In the event that the operator identifies a reactor trip is imminent and successfully initiates a manual reactor trip before the automatic 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. If manual reactor trip actions at the Control Room panels (following an unsuccessful automatic reactor trip) fail to reduce reactor power to or below 5%, the event escalates to the Site Area Emergency under EAL SS3.1.

CCNPP Basis Reference(s):

1. Technical Specifications 3.3.1, Reactor Protective System (RPS) Instrumentation - Operating
2. Technical Specifications 3.3.2, Reactor Protective System (RPS) Instrumentation - Shutdown
3. Technical Specifications 3.3.3, Reactor Protective System (RPS) Logic and Trip Initiation
4. EOP-0 Post-Trip Immediate Actions
5. UFSAR Section 7
6. AOP-3G Malfunction of Main Feedwater System
7. UFSAR 14.1.2.2.e
8. UFSAR 14.4.1 & Table 14.1-2
9. NEI 99-01 SA2

Category: S – System Malfunction

Subcategory: 3 – Criticality & RPS Failure

Initiating Condition: Automatic trip and manual actions taken from the reactor control console failed to shut down the reactor

EAL:

SS3.1 Site Area Emergency

An automatic reactor trip failed to shut down the reactor as indicated by reactor power > 5%

AND

Manual actions taken at the Control Room panels do **not** shut down the reactor as indicated by reactor power > 5%

Mode Applicability:

1 - Power Operation

Basis:

Generic

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful. A Site Area Emergency is warranted because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS.

The reactor should be considered shutdown when it producing less heat than the maximum decay heat load for which the safety systems are designed (5% power).

Manual scram (trip) actions taken at the Control Room panels are any set of actions by the reactor operator(s) at which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

Manual trip actions are not considered successful if action away from the Control Room panels is required to trip the reactor. This EAL is still applicable even if actions taken away from the Control Room panels are successful in shutting the reactor down because the design limits of the fuel may have been exceeded or because of the gross failure of the Reactor Protection System to shutdown the plant.

Escalation of this event to a General Emergency would be due to a prolonged condition leading to an extreme challenge to either core-cooling or heat removal.

Plant-Specific

Following a successful reactor trip, nuclear power promptly drops to about six percent of the original power level and then decays to a level some 8 decades less at a startup rate (SUR) of about (-)1/3 DPM. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from an automatic reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation (NI) and a negative SUR as nuclear power drops into the source range.

The RPS setpoints listed in Figure S-1 should result in an automatic reactor trip:

Figure S-1 Automatic RPS Trips

REACTOR TRIP		
High Power Level	2/4	Variable
High Rate-of-Change of Power	2/4 below 15% Pwr	2.6 decade/min.
Low Reactor Coolant Flow	2/4 above 10^{-4}% Pwr	Variable
Low Steam Generator Pressure	2/4	670 psig
Low Steam Generator Water Level	2/4	10 in. below top of feed ring
High Pressurizer Pressure	2/4	2385 psig
Thermal Margin/Low Pressure	2/4 above 10^{-4}% Pwr	Variable
Loss of Load	2/4 above 15% Pwr	N/A
High Containment Pressure	2/4	4 psig
Axial Flux Offset	2/4	Variable
Thermal Margin/SG Press. Diff. Hi	2/4 above 10^{-4}% Pwr	135 psid

Per EOP-0, Post-Trip Immediate Actions, the operator ensures that the reactor has tripped by depressing one set of Manual Reactor Trip buttons immediately following any symptoms of a reactor trip. The symptoms include:

- Reactor Trip alarm
- Control Element Assembly (CEA) Circuit Breaker(s) Trip alarms
- Rapid Lowering in Reactor Power
- Protection Channel Trip alarm
- Reactor Protective System (RPS) Trip Bistable Lights lit

Following depression of the reactor trip buttons, the operator verifies that reactor power is decreasing. If these responses cannot be verified, as part of contingency actions, the operator is instructed to open the motor generator (MG) set feeder breakers that provide power to the Control Element Drive Mechanism (CEDM).

If reactor power is above 5%, the reactor is producing more heat than the Auxiliary Feedwater system and Atmospheric Dump Valves are designed to remove (ref. 7, 8, 9). Fast boration is thus required and there is an actual major failure of a system intended for protection of the public. The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat poses a direct threat to the Fuel Clad and RCS barriers and warrants declaration of a Site Area Emergency.

CCNPP Basis Reference(s):

1. Technical Specifications 3.3.1, Reactor Protective System (RPS) Instrumentation - Operating
2. Technical Specifications 3.3.2, Reactor Protective System (RPS) Instrumentation - Shutdown
3. Technical Specifications 3.3.3, Reactor Protective System (RPS) Logic and Trip Initiation
4. EOP-0 Post-Trip Immediate Actions
5. EOP-8 Functional Recovery
6. UFSAR Section 7
7. AOP-3G Malfunction of Main Feedwater System
8. UFSAR 14.1.2.2.e
9. UFSAR 14.4.1 & Table 14.1-2
10. NEI 99-01 SS2

Category: S – System Malfunction

Subcategory: 3 – Criticality & RPS Failure

Initiating Condition: Automatic trip and **all** manual actions fail to shut down the reactor and indication of an extreme challenge to the ability to cool the core exists

EAL:

SG3.1 General Emergency

An automatic reactor trip failed to shut down the reactor as indicated by reactor power > 5%

AND

All manual actions fail to shut down the reactor as indicated by reactor power > 5%

AND

ANY of the following exist or have occurred:

- CET readings > 700°F
- RCS pressure > PORV setpoint
- RCS subcooling < 25°F

Mode Applicability:

1 - Power Operation

Basis:

Generic

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful.

The reactor should be considered shutdown when it producing less heat than the maximum decay heat load for which the safety systems are designed (5% power). In the event either of these challenges exists at a time that the reactor has not been brought below the power associated with the safety system design a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier table declaration to permit maximum off-site intervention time.

Plant-Specific

Following a successful reactor trip, nuclear power promptly drops to about six percent of the original power level and then decays to a level some 8 decades less at a startup rate (SUR) of about (-)1/3 DPM. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from an automatic reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation (NI) and a negative SUR as nuclear power drops into the source range.

The RPS setpoints listed in Figure S-1 should result in an automatic reactor trip:

Figure S-1 Automatic RPS Trips

REACTOR TRIP		
High Power Level	2/4	Variable
High Rate-of-Change of Power	2/4 below 15% Pwr	2.6 decade/min.
Low Reactor Coolant Flow	2/4 above 10^{-4}% Pwr	Variable
Low Steam Generator Pressure	2/4	670 psig
Low Steam Generator Water Level	2/4	10 in. below top of feed ring
High Pressurizer Pressure	2/4	2385 psig
Thermal Margin/Low Pressure	2/4 above 10^{-4}% Pwr	Variable
Loss of Load	2/4 above 15% Pwr	N/A
High Containment Pressure	2/4	4 psig
Axial Flux Offset	2/4	Variable
Thermal Margin/SG Press. Diff. Hi	2/4 above 10^{-4}% Pwr	135 psid

Per EOP-0, Post-Trip Immediate Actions, the operator ensures that the reactor has tripped by depressing one set of Manual Reactor Trip buttons immediately following any symptoms of a reactor trip. The symptoms include:

- Reactor Trip alarm
- Control Element Assembly (CEA) Circuit Breaker(s) Trip alarms
- Rapid Lowering in Reactor Power
- Protection Channel Trip alarm
- Reactor Protective System (RPS) Trip Bistable Lights lit

Following depression of the reactor trip buttons, the operator verifies that reactor power is decreasing. If these responses cannot be verified, as part of contingency actions, the operator is instructed to open the motor generator (MG) set feeder breakers that provide power to the Control Element Drive Mechanism (CEDM).

If reactor power is above 5%, the reactor is producing more heat than the Auxiliary Feedwater system and Atmospheric Dump Valves are designed to remove (ref. 7, 8, 9). Fast boration is thus required and there is an actual major failure of a system intended for protection of the public. The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat poses a direct threat to the Fuel Clad and RCS barriers.

Core Exit Thermocouples (CETs) are a component of the Inadequate Core Cooling Instrumentation system and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. The RCS Pressure Safety Limit is 2750 psia per CCNPP Technical Specifications. The saturation temperature for this pressure is 682.2°F. Per Action Value Bases Document EOP-24.33, the uncertainty on CET Temperature is +/- 39.8°F. If one or more CETs indicate 722°F (682.2 + 39.8), subcooling has been lost for at least some locations in the core. CET indications at or above 722°F are a clear sign that core heat removal capability is lost or greatly reduced and one fission product barrier, the fuel clad, is threatened due to elevated fuel temperatures. 700°F qualifies as a condition representing a potential loss of the fuel clad barrier (ref. 13).

Inability to remove heat from the RCS to the ultimate heat sink (bay or atmosphere) is a loss of function required for hot shutdown with the reactor at pressure and temperature and thus represents potential loss of the Fuel Clad and RCS barriers.

The combination of these conditions (reactor power greater than 5% with loss of subcooling margin or inability to remove heat from the RCS) indicates the ultimate heat sink function is under extreme challenge, a core melt sequence may exist and rapid degradation of the fuel clad could begin. To permit maximum offsite intervention time, the General Emergency declaration is appropriate in anticipation of an inevitable General Emergency declaration due to loss and potential loss of fission product barriers.

CCNPP Basis Reference(s):

1. Technical Specifications 3.3.1, Reactor Protective System (RPS) Instrumentation - Operating
2. Technical Specifications 3.3.2, Reactor Protective System (RPS) Instrumentation - Shutdown
3. Technical Specifications 3.3.3, Reactor Protective System (RPS) Logic and Trip Initiation
4. EOP-0 Post-Trip Immediate Actions
5. EOP-8 Functional Recovery
6. UFSAR Section 7
7. AOP-3G Malfunction of Main Feedwater System
8. UFSAR 14.1.2.2.e
9. UFSAR 14.4.1 & Table 14.1-2
10. ERPIP-800 Core Damage Assessment
11. ERPIP-802 Core Damage Assessment Using Core Exit Thermocouples
12. EOP-5 Loss of Coolant Accident
13. EOP-24.33 Action Value Bases Document
14. CEN-152 Emergency Procedure Guidelines
15. OP-7 Shutdown Operations
15. ERPIP-601 Severe Accident Management Initial Diagnosis
16. NEI 99-01 SG2

Category: S – System Malfunction

Subcategory: 4 – Inability to Reach or Maintain Shutdown Conditions

Initiating Condition: Inability to reach required shutdown within Technical Specification limits

EAL:

SU4.1	**Unusual Event**
Plant is **not** brought to required operating mode within Technical Specifications LCO required action completion time	

Mode Applicability:

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

Basis:

Generic

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required operating mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the site Technical Specifications requires a four hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable required action completion time in the Technical Specifications. An immediate UE is required when the plant is not brought to the required operating mode within the allowable required action completion time in the Technical Specifications. Declaration of a UE is based on the time at which the LCO-specified required action completion time period elapses under the site Technical Specifications and is not related to how long a condition may have existed.

Plant-Specific

None

CCNPP Basis Reference(s):

1. Technical Specifications 3.0, Limiting Conditions for Operations (LCO) Applicability
2. NEI 99-01 SU2

Category: S – System Malfunction

Subcategory: 5 – Instrumentation

Initiating Condition: Unplanned loss of safety system annunciation or indication in the Control Room for ≥ 15 min.

EAL:

SU5.1	**Unusual Event**
UNPLANNED loss of greater than approximately 75% of safety system annunciation or indication on Control Room panels for ≥ 15 min. (Note 4)	

Note 4: The ED 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

Mode Applicability:

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

Basis:

Generic

This EAL is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer based indication equipment is considered.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions.

It is further recognized that plant design provides redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on EAL SU4.1.

Annunciators or indicators for this EAL include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

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

This UE will be escalated to an Alert based on a concurrent loss of compensatory indications or if a significant transient is in progress during the loss of annunciation or indication.

Plant-Specific

The Control Room Panels that house safety related annunciators are listed in the table below:

Unit 1	Unit 2
1C04	2C04
1C05	2C05
1C06	2C06
1C07	2C07
1C08	2C08
1C09	2C09
1C10	2C10
1C13	2C13
1C18A	1C19C
1C18B	1C20
1C19C	1C20A
1C22	1C20B
1C24B	1C22
1C26	1C24B
1C33	1C26
1C34	1C33
	1C34

Definitions:

Unplanned

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

CCNPP Basis Reference(s):

1. UFSAR Sections 7.6 and 7.7
2. AOP-7J Loss of 120 Volt Vital AC or 125 Volt Vital DC Power
3. UFSAR 7.5.5
4. OI-50A Plant Computer
5. OP-AA-103-102, Watch Standing Practices
6. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 5 – Instrumentation

Initiating Condition: Unplanned loss of safety system annunciation or indication in the Control Room with either (1) a significant transient in progress, or (2) compensatory indicators are unavailable

EAL:

SA5.1 Alert

UNPLANNED loss of greater than approximately 75% of safety system annunciation or indication on Control Room panels for ≥ 15 min. (Note 4)

AND EITHER:

A significant transient is in progress, Table S-2

OR

Compensatory indications are unavailable (Plant Computer, SPDS)

Note 4: The ED 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

Table S-2 Significant Transients
• Automatic turbine runback > 25% thermal power
• Electric load rejection > 25% full electrical load
• Reactor trip
• Safety Injection actuation

Mode Applicability:

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

Basis:

Generic

This EAL is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a significant transient.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on EAL SU4.1.

Annunciators or indicators for this EAL include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

"Compensatory indications" in this context includes computer based information such as Plant Process Computer and SPDS.

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

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress due to a concurrent loss of compensatory indications with a significant transient in progress during the loss of annunciation or indication.

Plant-Specific

Plant Process Computer and SPDS are considered compensatory indication.

Significant transients are listed in Table S-2.

The Control Room Panels that house safety related annunciators are listed in the table below:

Unit 1	Unit 2
1C04	2C04
1C05	2C05
1C06	2C06
1C07	2C07
1C08	2C08
1C09	2C09
1C10	2C10
1C13	2C13
1C18A	1C19C

1C18B	1C20
1C19C	1C20A
1C22	1C20B
1C24B	1C22
1C26	1C24B
1C33	1C26
1C34	1C33
	1C34

Definitions:

Unplanned

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

CCNPP Basis Reference(s):

1. UFSAR Sections 7.6 and 7.7
2. AOP-7J Loss of 120 Volt Vital AC or 125 Volt Vital DC Power
3. UFSAR 7.5.5
4. OI-50A Plant Computer
5. OP-AA-103-102, Watch Standing Practices
6. NEI 99-01 SA4

Category: S – System Malfunction

Subcategory: 5 – Instrumentation

Initiating Condition: Inability to monitor a significant transient in progress

EAL:

SS5.1 Site Area Emergency

Loss of greater than approximately 75% of safety system annunciation or indication on Control Room panels for ≥ 15 min. (Note 4)

AND

A significant transient is in progress, Table S-2

AND

Compensatory indications are unavailable (Plant Computer, SPDS)

Note 4: The ED 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

Table S-2 Significant Transients
• Automatic turbine runback > 25% thermal power • Electric load rejection > 25% full electrical load • Reactor trip • Safety Injection actuation

Mode Applicability:

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

Basis:

Generic

This EAL is intended to recognize the threat to plant safety associated with the complete loss of capability of the Control Room staff to monitor plant response to a significant transient.

"Planned" and "unplanned" actions are not differentiated since the loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the NOUE is based on EAL SU4.1

A Site Area Emergency is considered to exist if the Control Room staff cannot monitor safety functions needed for protection of the public while a significant transient is in progress.

Annunciators for this EAL are limited to include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g. area, process, and/or effluent rad monitors, etc.)

Indications needed to monitor safety functions necessary for protection of the public include Control Room indications, computer generated indications and dedicated annunciation capability.

"Compensatory indications" in this context includes computer based information such as Plant Process Computer and SPDS.

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

Plant-Specific

Plant computer and SPDS are considered compensatory indication.

Significant transients are listed in Table S-2.

The Control Room Panels that house safety related annunciators are listed in the table below:

Unit 1	Unit 2
1C04	2C04
1C05	2C05
1C06	2C06
1C07	2C07
1C08	2C08
1C09	2C09
1C10	2C10
1C13	2C13

1C18A	1C19C
1C18B	1C20
1C19C	1C20A
1C22	1C20B
1C24B	1C22
1C26	1C24B
1C33	1C26
1C34	1C33
	1C34

CCNPP Basis Reference(s):

1. UFSAR Sections 7.6 and 7.7
2. AOP-7J Loss of 120 Volt Vital AC or 125 Volt Vital DC Power
3. UFSAR 7.5.5
4. OI-50A Plant Computer
5. CNG-OP-1.01-2003 Alarm Response and Control
6. NEI 99-01 SS6

Category: S – System Malfunction

Subcategory: 6 – Communications

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

EAL:

SU6.1 Unusual Event

Loss of **all** Table S-3 onsite (internal) communication methods affecting the ability to perform routine operations

OR

Loss of **all** Table S-3 offsite (external) communication methods affecting the ability to perform offsite notifications to any agency

Table S-3 Communications Systems		
System	**Onsite** (internal)	**Offsite** (external)
Commercial phone system	X	X
Plant page system	X	
FTS 2001 telephone system		X
CCNPP Radio System	X	
Satellite Phone System		X
Cellular Phone System	X	X

Mode Applicability:

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

Basis:

Generic

The purpose of this EAL is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate issues with off-site authorities.

The loss of off-site communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary off-site communications is sufficient to inform federal, state, and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from non-routine radio transmissions, individuals being sent to off-site locations, etc.) are being used to make communications possible.

Plant-Specific

Onsite/offsite communications systems are listed in Table S-3 (ref. 1, 2, 3).

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

CCNPP Basis Reference(s):

1. UFSAR Section 7.8
2. Emergency Response Facility Directory & Communications Equipment Information
3. NO-1-113, Control of Radio Transmitter (PRT)
4. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 7 – Fuel Clad Degradation

Initiating Condition: Fuel clad degradation

EAL:

SU7.1	**Unusual Event**

Coolant activity > **ANY** of the following:

- Dose equivalent I-131 0.5 uCi/gm for 100 hrs. continuous
- Dose equivalent I-131 acceptable region of T.S. Fig. 3.4.15-1
- Dose equivalent I-131 137.5 uCi/gm
- Gross activity 100/E-bar uCi/gm

Mode Applicability:

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

Basis:

Generic

This EAL is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

Escalation of this EAL to the Alert level is via the EALs in Category F.

This threshold addresses coolant samples exceeding coolant technical specifications limits.

Plant-Specific

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. This EAL addresses reactor coolant samples exceeding coolant technical specifications (including allowable transient time limits permitted in the Technical Specifications). Though the referenced Technical Specification limits are mode dependent, it is appropriate that the EAL be applicable in modes 1-4, as it indicates a potential degradation in the level of safety of the plant.

For mode 4, conditions are not governed by the Technical Specification applicability. While in mode 4 however, any abnormal chemistry sample that is reported will be compared to the criteria of this EAL and an Unusual Event will be declared if met.

The maximum value shown on T.S. Figure 3.4. 15-1 is 137.5 uCi/gm dose equivalent I-131; therefore this value is included in the EAL threshold to address plant conditions below that shown in the T.S. figure. (ref. 1, 2).

CCNPP Basis Reference(s):

1. Technical Specification 3.4.15 Reactor Coolant System - RCS Specific Activity
2. AOP-6A Abnormal Reactor Coolant Chemistry/Activity
3. 1(2)C07-ALM F-21 RAD MON LVL HI
4. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 7 – Fuel Clad Degradation

Initiating Condition: Fuel clad degradation

EAL:

SU7.2	**Unusual Event**
Letdown Monitor (RY-202-1) high alarm (≥ 1E+06 cpm)	

Mode Applicability:

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

Basis:

Generic

This EAL is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

Escalation of this EAL to the Alert level is via the EALs in Category F.

This threshold addresses radiation monitor readings that provide indication of a degradation of fuel clad integrity.

Plant-Specific

This EAL addresses indication of gross failed fuel that may be in excess of Technical Specification (ref. 1) coolant activity limits.

The Letdown Radiation Monitor (1(2)-RY-202-1)) gross radiation channel continuously monitors the activity in a sample drawn from the RCS and actuates an alarm in the Control Room if a predetermined activity level is reached (ref. 3). The sensor is a gross-gamma plus specific isotope (I-135) monitor; the system is designed to detect activity release from the fuel to the reactor coolant within five minutes of the event (ref. 2, 3). The instrument range is 10 - 1E+6 cpm. RY-202-1 does not read out in µCi/cc (ref. 4).

CCNPP Basis Reference(s):

1. Technical Specification 3.4.15 Reactor Coolant System - RCS Specific Activity
2. AOP-6A Abnormal Reactor Coolant Chemistry/Activity
3. 1(2)C07-ALM F-21 RAD MON LVL HI
4. UFSAR Section 9.1.3
5. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 8 – RCS Leakage

Initiating Condition: RCS leakage

EAL:

SU8.1	**Unusual Event**
Unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. (Note 4)	
OR	
Identified leakage > 25 gpm for ≥ 15 min. (Note 4)	

Note 4: The ED 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.

Mode Applicability:

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

Basis:

Generic

This EAL is included as a UE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified or pressure boundary leakage was selected as it is observable with normal Control Room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances).

Relief valve normal operation should be excluded from this EAL. However, a relief valve that operates and fails to close per design should be considered applicable to this EAL if the relief valve cannot be isolated.

The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. In either case, escalation of this EAL to the Alert level is via EALs in Category F.

Plant-Specific

STP 0-27-1(2), RCS Leakage Evaluation, provides instructions for calculating primary system leak rate by manual or PC program methods (ref. 2). AOP-2A-1(2), Excessive Reactor Coolant Leakage, provides direction for determining RCS leakage for off normal events and for operations troubleshooting (ref. 3).

Calvert Cliffs Technical Specifications do not treat Steam Generator tube leakage as RCS pressure boundary leakage. Since Steam Generator tube leakage is identified leakage, the threshold for Unususal Event is > 25 gpm under this initiating condition.

CCNPP Basis Reference(s):

1. Technical Specifications 3.4.13, Reactor Coolant System Operational Leakage
2. AOP-2A Excessive Reactor Coolant Leakage
3. STP 0-27-1(2) RCS Leakage Evaluation
4. Technical Specifications 1.1, Definitions
5. NEI 99-01 SU5

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. Reactor Fuel Clad (FC): The fuel clad barrier consists of fuel bundle tubes composed of zirconium-based alloys that contain the fuel pellets.

B. Reactor Coolant System (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 (CNMT): 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.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 2). "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures Containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Unusual Event:
Any loss or any potential loss of Containment

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

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

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

The logic used for Category F EALs reflects the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier. UE EALs associated with RCS and Fuel Clad Barriers are addressed under Category S.
- At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier "Loss" thresholds existed, that, in addition to off-site dose assessments, would require continual assessments of radioactive inventory and Containment integrity. Alternatively, if both Fuel Clad and RCS Barrier "Potential Loss" thresholds existed, the ED would have more assurance that there was no immediate need to escalate to a General Emergency.
- The ability to escalate to higher emergency classification levels as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.
- The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment Barrier status is addressed by Technical Specifications.

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **ANY** loss or **ANY** potential loss of Containment

EAL:

FU1.1	**Unusual Event**
ANY loss or **ANY** potential loss of Containment (Table F-1)	

Mode Applicability:

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

Basis:

Generic

None

Plant-Specific

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Fuel Clad and RCS barriers, the loss of either of which results in an Alert (EAL FA1.1), loss of the Containment barrier in and of itself does not result in the relocation of radioactive materials or the potential for degradation of core cooling capability. However, loss or potential loss of the Containment barrier in combination with the loss or potential loss of either the Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1.

CCNPP Basis Reference(s):

1. NEI 99-01 FU1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **ANY** loss or **ANY** potential loss of either Fuel Clad or RCS

EAL:

FA1.1 Alert
ANY loss or **ANY** potential loss of either Fuel Clad or RCS (Table F-1)

Mode Applicability:

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

Basis:

Generic

None

Plant-Specific

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the 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.

CCNPP Basis Reference(s):

1. NEI 99-01 FA1

Category: Fission Product Barrier Degradation

Subcategory: N/A

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

EAL:

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

Mode Applicability:

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

Basis:

Generic

None

Plant-Specific

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

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

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

At the Site Area Emergency classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, the Emergency Director would have greater assurance that escalation to a General Emergency is less imminent.

CCNPP Basis Reference(s):

1. NEI 99-01 FS1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss of **ANY** two barriers and loss or potential loss of the third barrier

EAL:

FG1.1	**General Emergency**
Loss of **ANY** two barriers **AND** Loss or potential loss of the third barrier (Table F-1)	

Mode Applicability:

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

Basis:

Generic

None

Plant-Specific

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

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

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

CCNPP Basis Reference(s):

1. NEI 99-01 FG1

ATTACHMENT 2

Fission Product Barrier Loss / Potential Loss Matrix and Basis

Introduction

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

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

A. Core Cooling / Heat Removal

B. Inventory

C. Radiation / Coolant Activity

D. Isolation Status

E. Judgment

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

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A is "FC Loss A.1," the third Containment barrier Potential Loss is "CNMT P-Loss B.3," etc.

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

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

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

The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if Containment radiation is sufficiently high (i.e., greater than 14,000 R/hr), a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Containment barrier exist. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, FA1.1 and FU1.1 to determine the appropriate emergency classification.

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

Table F-1 Fission Product Barrier Matrix						
	Fuel Clad Barrier		Reactor Coolant System Barrier		Containment Barrier	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A Core Cooling / Heat Removal	1. CET readings > 1,200°F	1. CET readings > 700°F 2. RCS heat removal **cannot** be established **AND EITHER**: RCS pressure > PORV setpoint **OR** RCS subcooling < 25°F	None	1. OTCC flow established 2. RCS heat removal **cannot** be established **AND EITHER**: RCS pressure > PORV setpoint **OR** RCS subcooling < 25°F 3. Uncontrolled RCS cooldown and to left of Max Operating Pressure Curve (EOP Attachment 1, RCS Pressure Temperature Limits)	None	1. CET readings **cannot** be restored < 1,200°F within 15 min. 2. CET readings > 700°F **AND** Reactor vessel water level **cannot** be restored > RVLMS 10 in. alarm within 15 min.
B Inventory	None	3. RVLMS level ≤ 10 in. alarm	1. RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling (< 25°F) 2. RUPTURED S/G results in an ECCS (SIAS) actuation	4. RCS leak rate > 50 gpm with letdown isolated	1. A Containment pressure rise followed by a rapid unexplained drop in Containment pressure 2. Containment pressure or sump level response **not** consistent with LOCA conditions 3. RUPTURED S/G (> 50 gpm) is also FAULTED outside of Containment 4. Primary-to-secondary leakrate > 10 gpm **AND** Unisolable prolonged steam release from affected S/G to the environment	3. Containment pressure ≥ 50 psig and rising 4. Containment hydrogen concentration ≥ 4% 5. Containment pressure > 4.25 psig **AND cannot** meet **ANY** of the following conditions: • 2 Containment Spray Pumps Operating • 3 CACs Operating • 1 Containment Spray Pump and 2 CACs Operating
C Radiation / Coolant Activity	2. Containment radiation monitor (5317A/B) reading > 3,500 R/hr 3. Post-accident sample dose rate ≥ 40 mRem/hr (1 ft from sample) 4. Coolant activity >300 µCi/cc DEQ I-131	None	3. Containment radiation monitor (5317A/B) reading > 6 R/hr (Note 8)	None	None	6. Containment radiation monitor (5317A/B) reading > 14,000 R/hr
D Isolation Status	None	None	None	None	5. Failure of **all** valves in **ANY one** line to close **AND** Direct downstream pathway to the environment exists after Containment isolation signal	None
E Judgment	5. **ANY** condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier	4. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier	4. **ANY** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	5. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the Containment barrier	7. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier

Barrier: Fuel Clad

Category: A. Core Cooling/Heat Removal

Degradation Threat: Loss

Threshold:

1. CET readings > 1,200°F

Basis:

Generic

The 1,200°F reading corresponds to significant superheating of the coolant.

Plant-Specific

Core Exit Thermocouples (CETs) are a component of Inadequate Core Cooling Instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. The threshold temperature is consistent with Attachment 3 of ERPIP-802, Core Damage Assessment Using Core Exit Thermocouples. Although clad rupture due to high temperature is not expected for CET readings less than the threshold, temperatures of this magnitude signal significant superheating of the reactor coolant and core uncovery. Events that result in CET readings above the loss threshold are classified severe accidents and lead to a Severe Accident Management Guideline "Badly Damaged (BD)" condition. The BD descriptor signifies possible core overheating to the point clad ballooning/collapse may have occurred and portions of the core may melt.

CCNPP Basis Reference(s):

1. UFSAR 7.5.9
2. ERPIP-600 Severe Accident Management
3. ERPIP-601 Severe Accident Management Initial Diagnosis
4. ERPIP-802 Core Damage Assessment Using Core Exit Thermocouples
5. EOP-5 Loss of Coolant Accident
6. EOP-8 Functional Recovery Procedure
7. CEN-152 Emergency Procedure Guidelines

Barrier: Fuel Clad

Category: A. Core Cooling/Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CET readings > 700°F

Basis:

Generic

CET readings > 700°F corresponds to loss of subcooling.

Plant-Specific

Core Exit Thermocouples (CETs) are a component of the Inadequate Core Cooling Instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. The RCS Pressure Safety Limit is 2750 psia per CCNPP Technical Specifications. The saturation temperature for this pressure is 682.2°F. Per Action Value Bases Document EOP-24.33, the uncertainty on CET Temperature is +/- 39.8°F. If one or more CETs indicate 722°F (682.2 + 39.8), subcooling has been lost for at least some locations in the core. CET indications at or above 722°F are a clear sign that core heat removal capability is lost or greatly reduced and one fission product barrier, the fuel clad, is threatened due to elevated fuel temperatures. 700°F qualifies as a condition representing a potential loss of the fuel clad barrier (ref. 6).

CCNPP Basis Reference(s):

1. UFSAR 7.5.9
2. ERPIP-800 Core Damage Assessment
3. ERPIP-802 Core Damage Assessment Using Core Exit Thermocouples
4. EOP-5 Loss of Coolant Accident
5. EOP-8 Functional Recovery Procedure
6. EOP-24.33 Action Value Bases Document
7. CEN-152 Emergency Procedure Guidelines

Barrier: Fuel Clad

Category: A. Core Cooling/Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. RCS heat removal **cannot** be established

   AND EITHER:

   RCS pressure > PORV setpoint

   OR

   RCS subcooling < 25°F

Basis:

Generic

This subcategory addresses other site specific thresholds that may be included to indicate loss or potential loss of the Fuel Clad barrier.

Plant-Specific

The steam generators (S/Gs) provide the normal means of heat transfer from the RCS to the main condenser and ultimate heat sink. EOP-5, Loss of Coolant Accident, requires maintenance of S/G heat removal at all times during a LOCA, if at all possible. Once RCS pressure and temperature are reduced, RCS heat removal can be provided by Shutdown Cooling (SDC). Once the SDC is placed in service, the S/G heat sink capability is no longer necessary.

S/Gs are available for RCS heat removal if the level in at least one S/G can be restored and maintained above -170 in. and T_{COLD} is not increasing. Core and RCS heat removal is available if CET readings are less than superheated and the temperature difference between hot leg temperature and cold leg temperature is less than 50°F (10°F with forced circulation). If RCS pressure approaches the PORV setpoint (2,400 psia), heat input to the RCS is likely raising pressure instead of reaching the ultimate heat sink. If RCS subcooling approaches 25°F, the margin to superheated conditions is being reduced. Following an uncomplicated reactor trip, subcooling margin should be in excess of 50°F. Subcooling margin greater than 25°F ensures the fluid surrounding the core is sufficiently cooled and provides margin for reestablishing flow should subcooling deteriorate when SI flow is secured. Voids may exist in some parts of the RCS (e.g., Reactor Vessel head) but are permissible as long as core heat removal is maintained.

The combination of these conditions indicates the ultimate heat sink function is under extreme challenge. This threshold addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a potential loss of the Fuel Clad barrier. This is also a potential loss of the RCS barrier and therefore results in at least a Site Area Emergency.

CCNPP Basis Reference(s):

1. UFSAR Section 7.5.9
2. OP-7 Shutdown Operations
3. ERPIP-601 Severe Accident Management Initial Diagnosis
4. ERPIP-800 Core Damage Assessment
5. ERPIP 802 Core Damage Assessment Using Core Exit Thermocouples
6. EOP-5 Loss of Coolant Accident
7. EOP-8 Functional Recovery Procedure
8. EOP-23.02 Subcooling Margin (SCM): 25 Deg F Subcooled
9. CEN-152 Emergency Procedure Guidelines

Barrier: Fuel Clad

Category: B. Inventory

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: B. Inventory

Degradation Threat: Potential Loss

Threshold:

3. RVLMS < 10 in. alarm

Basis:

Generic

There is no Loss threshold associated with this item.

The site specific value for the Potential Loss threshold corresponds to the top of the active fuel.

Plant-Specific

The Reactor Vessel Level Monitoring System (RVLMS) is based on the CE Heated Junction Thermocouple (HJTC) system. The HJTC system measures reactor coolant liquid inventory with discrete HJTC sensors located at different levels within a separator tube ranging from the fuel alignment plate (i.e., near top of active fuel) to the top of the Reactor Vessel head. The basic principle of system operation is detection of a temperature difference between heated and unheated thermocouples.

Reactor Vessel water level below the top of the core may lead to a Severe Accident Management Guideline "Badly Damaged (BD)" condition. The BD descriptor signifies possible core overheating to the point of clad ballooning/collapse and melting. When Reactor Vessel/RCS water level drops to 32.9 ft el., core uncovery is about to occur. The closest RVLMS indication is the 10 in. alarm. This signals inadequate coolant inventory, loss of subcooling and the occurrence of possible fuel clad damage.

CCNPP Basis Reference(s):

1. UFSAR 7.5.9
2. ERPIP-800 Core Damage Assessment
3. OP-7 Shutdown Operations
4. ERPIP-600 Severe Accident Management
5. ERPIP-601 Severe Accident Management Initial Diagnosis

Barrier: Fuel Clad

Category: C. Radiation / Coolant Activity

Degradation Threat: Loss

Threshold:

2. Containment radiation monitor (5317A/B) reading > 3,500 R/hr

Basis:

Generic

The 3,500 R/hr Containment radiation monitor reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the Containment.

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.

This value is higher than that specified for RCS barrier Loss threshold #3. Thus, this threshold indicates a loss of both the Fuel Clad barrier and RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

There is no Potential Loss threshold associated with this item.

Plant-Specific

Containment radiation is indicated on 1(2)-RI-5317 A&B.

CCNPP Basis Reference(s):

1. ERPIP-801 Core Damage Assessment Using Containment Radiation Dose Rates

Barrier: Fuel Clad

Category: C. Radiation / Coolant Activity

Degradation Threat: Loss

Threshold:

3. Post-accident sample dose rate ≥ 40 mRem/hr (1 ft from sample)

Basis:

Generic

The post accident sample dose rate value corresponds to 300 μCi/gm I-131 equivalent. Assessment by the EAL Task Force indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost.

There is no Potential Loss threshold associated with this item.

Plant-Specific

A shielded 12.5 ml pressurized bomb sample would read 40 mRem/hr at one foot from the sample (168 mRem/hr unshielded) for 5% fuel clad damage. When reactor coolant activity reaches this level, significant clad heating has occurred and thus the Fuel Clad barrier is considered lost (ref. 1).

CCNPP Basis Reference(s):

1. BG&E Fuel Degradation EALs Calculation Worksheet, JSB Associates, February 18, 1993

Barrier: Fuel Clad

Category: C. Radiation / Coolant Activity

Degradation Threat: Loss

Threshold:

4. Coolant activity > 300 μCi/cc DEQ I-131

Basis:

Generic

The site specific value corresponds to 300 μCi/cc I-131 equivalent. Assessment by the EAL Task Force indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost.

There is no Potential Loss threshold associated with this item.

Plant-Specific

None

CCNPP Basis Reference(s):

1. NEI 99-01 Revision 5, pg 35

Barrier: Fuel Clad

Category: C. Radiation / Coolant Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. Isolation Status

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. Isolation Status

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. Judgment

Degradation Threat: Loss

Threshold:

5. **ANY** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier

Basis:

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

CCNPP Basis Reference(s):

None

Barrier: Fuel Clad

Category: E. Judgment

Degradation Threat: Potential Loss

Threshold:

4. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier

Basis:

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is I potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

CCNPP Basis Reference(s):

None

Barrier: Reactor Coolant System

Category: A. Core Cooling/Heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: A. Core Cooling/Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. OTCC flow established

Basis:

Generic

This subcategory addresses other site specific thresholds that may be included to indicate loss or potential loss of the RCS barrier.

Plant-Specific

CCNPP is a CE plant with Once Through Core Cooling (OTCC) capability and has a procedure that intentionally opens the RCS barrier to cool the core when normal means fail. This procedure is employed when the heat removal function is extremely challenged. Establishment of OTCC flow represents a potential loss of the RCS barrier due to PORVs being intentionally maintained open to establish adequate core heat removal capability.

CCNPP Basis Reference(s):

1. EOP-3 Loss of All Feedwater

Barrier: Reactor Coolant System

Category: A. Core Cooling/Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. RCS heat removal **cannot** be established

   AND EITHER:

   RCS pressure > PORV setpoint

   OR

   RCS subcooling < 25°F

Basis:

Generic

This subcategory addresses other site specific thresholds that may be included to indicate loss or potential loss of the RCS barrier.

Plant-Specific

The steam generators (S/Gs) provide the normal means of heat transfer from the RCS to the main condenser and ultimate heat sink. By cooling down the S/Gs, heat input to the RCS from the reactor core is reduced. EOP-5, Loss of Coolant Accident, requires maintenance of S/G heat removal at all times during a LOCA, if at all possible. Once RCS pressure and temperature are reduced, RCS heat removal can be provided by the Shutdown Cooling (SDC). Once the SDC is placed in service, the S/G heat sink capability is no longer necessary.

S/Gs are available for RCS heat removal if the level in at least one S/G can be restored and maintained above -170 in. and T_{COLD} is not increasing. Core and RCS heat removal is available if CET readings are less than superheated and the temperature difference between hot leg temperature and cold leg temperature is less than 50°F (10°F with forced circulation). If RCS pressure approaches the PORV setpoint (2400 psia), heat input to the RCS is likely raising pressure instead of reaching the ultimate heat sink. If RCS subcooling approaches 25°F, the margin to superheated conditions is being reduced. Following an uncomplicated reactor trip, subcooling margin should be in excess of 50°F. Subcooling margin greater than 25°F ensures the fluid surrounding the core is sufficiently cooled and provides margin for reestablishing flow should subcooling deteriorate when SI flow is secured. Voids may exist in some parts of the RCS (e.g., Reactor Vessel head) but are permissible as long as core heat removal is maintained.

The combination of these conditions indicates the ultimate heat sink function is under extreme challenge. This threshold addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a potential loss of the RCS barrier. This is also a potential loss of the Fuel Clad barrier and therefore results in at least a Site Area Emergency.

CCNPP Basis Reference(s):

1. UFSAR Section 7.5.9
2. OP-7 Shutdown Operations
3. ERPIP-601 Severe Accident Management Initial Diagnosis
4. ERPIP-800 Core Damage Assessment
5. ERPIP 802 Core Damage Assessment Using Core Exit Thermocouples
6. EOP-5 Loss of Coolant Accident
7. EOP-8 Functional Recovery Procedure
8. EOP-23.02 Subcooling Margin (SCM): 25 Deg F Subcooled
9. CEN-152 Emergency Procedure Guidelines

Barrier: Reactor Coolant System

Category: A. Core Cooling/Heat Removal

Degradation Threat: Potential Loss

Threshold:

3. Uncontrolled RCS cooldown and to left of Max Operating Pressure Curve (EOP Attachment 1, RCS Pressure Temperature Limits)

Basis:

Generic

This subcategory addresses other site specific thresholds that may be included to indicate loss or potential loss of the RCS barrier.

Plant-Specific

"Uncontrolled" means that the RCS cooldown was not the result of deliberate action performed in accordance with plant procedures and exceeds allowable vessel cooldown limits. Among the EOP safety functions to be maintained is RCS Pressure Control. Per EOP-4, Excess Steam Demand Event, the potential exists for pressurized thermal shock from an excessive cooldown rate followed by a repressurization.

The Max Operating Pressure Curve and RCS cooldown rate limits are established to prevent the effects of pressurized thermal shock. The region to the left of the curve is labeled the "Non-Operating Area." Several curves are included in EOP Attachment 1 based on the combinations of Reactor Coolant Pumps (RCPs) in operation. The combination of the conditions of this potential loss threshold indicates the RCS barrier is under significant challenge.

CCNPP Basis Reference(s):

1. EOP-4 Excessive Steam Demand Event
2. EOP-Attachments, Attachment 1 RCS Pressure Temperature Limits
3. EOP-Attachments, Attachment 14 RCS Cooldown Data Sheet
4. EOP-8 Functional Recovery Procedure

Barrier: Reactor Coolant System

Category: B. Inventory

Degradation Threat: Loss

Threshold:

1. RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling (< 25°F)

Basis:

Generic

This threshold addresses conditions where leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak.

Plant-Specific

AOP-2A, Excessive Reactor Coolant Leakage, provides a list of conditions that may be observed when excessive RCS leakage occurs and provides appropriate actions to prevent and mitigate the consequences of RCS leakage. Following an uncomplicated reactor trip, subcooling margin should be in the range of 50°F to 75°F. Subcooling margin greater than or equal to 25°F ensures the fluid surrounding the core is sufficiently cooled and provides margin for reestablishing flow should subcooling deteriorate when SIS flow is secured. Voids may exist in some parts of the RCS (e.g., Reactor Vessel head) but are permissible as long as core heat removal is maintained. The loss of subcooling is therefore the fundamental indication that the inventory control systems are incapable of counteracting the mass loss through the leak in the RCS.

The loss of subcooling as a result of inability to establish RCS heat transfer to the ultimate heat sink is indicative of potential losses of the Fuel Clad and RCS barriers.

CCNPP Basis Reference(s):

1. EOP-8 Subcooling Margin (SCM): 25 Deg F Subcooled
2. AOP-2A Excessive Reactor Coolant Leakage
3. EOP-5 Loss of Coolant Accident
4. EOP-6 Steam Generator Tube Rupture

Barrier: Reactor Coolant System

Category: B. Inventory

Degradation Threat: Loss

Threshold:

2. RUPTURED S/G results in an ECCS (SIAS) actuation

Basis:

Generic

This threshold addresses the full spectrum of Steam Generator (SG) tube rupture events in conjunction with Containment barrier loss thresholds. It addresses ruptured SG(s) for which the leakage is large enough to cause actuation of ECCS (SIAS). This is consistent to the RCS leak rate barrier potential loss threshold.

By itself, this threshold will result in the declaration of an Alert. However, if the SG is also faulted (i.e., two barriers failed), the declaration escalates to a Site Area Emergency per Containment barrier loss thresholds.

There is no potential loss threshold associated with this item.

Plant-Specific

Definitions:

Ruptured

In a steam generator, existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

CCNPP Basis Reference(s):

1. EOP-5 Loss of Coolant Accident EOP-5 Loss of Coolant Accident
2. EOP-6 Steam Generator Tube Rupture
3. 1C08-ALM ESFAS 11, G-05
4. Technical Specifications Table 3.3.4-1

Barrier: Reactor Coolant System

Category: B. Inventory

Degradation Threat: Potential Loss

Threshold:

4. RCS leak rate > 50 gpm with letdown isolated

Basis:

Generic

This threshold is based on the apparent inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Chemical and Volume Control System which is considered to be the flow rate equivalent to one charging pump discharging to the charging header. Isolating letdown is a standard abnormal operating procedure action and may prevent unnecessary classifications when a non-RCS leakage path such as a CVCS leak exists. The intent of this condition is met if attempts to isolate Letdown are NOT successful. Additional charging pumps being required is indicative of a substantial RCS leak.

Plant-Specific

The CVCS includes three positive displacement horizontal pumps with a capacity of 44 gpm each. The pressurizer level control program regulates letdown purification subsystem flow by adjusting the letdown flow control valve so that the reactor coolant pump (RCP) controlled leak-off plus the letdown flow matches the input from the operating charging pump. Equilibrium pressurizer level conditions may be disturbed due to RCS temperature changes, power changes, or RCS inventory loss due to leakage. A decrease in pressurizer water level below the programmed level results in a control signal to start one or both standby charging pumps to restore water level. The need for a second or third charging pump to makeup leakage in excess of letdown flow would be indicative of substantial RCS leakage. The single charging pump capacity is rounded up to 50 gpm for this threshold and clearly signals that operation of more than one charging pump is needed.

CCNPP Basis Reference(s):

1. UFSAR Section 9.1.3

Barrier: Reactor Coolant System

Category: C. Radiation / Coolant Activity

Degradation Threat: Loss

Threshold:

3. Containment radiation monitor (5317A/B) reading > 6 R/hr (Note 8)

Note 8: High temperature in Containment may induce a current error in the Mineral Insulated (MI) cable running through Containment to the meter. The CHRRM 1(2)-RI-5317 A&B may **not** detect this value (6 R/hr) under these conditions. When Containment temperature reaches 300°F, the meter will indicate approximately 40 R/hr for a few minutes then drop to approximately 10 R/hr after three hours. This information is to provide guidance on determining the validity of the readings under the specified high temperature conditions.

Basis:

Generic

The site specific reading is a value which indicates the release of reactor coolant to the Containment.

This reading is less than that specified for Fuel Clad barrier threshold 2. Thus, this threshold would be indicative of a RCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad barrier threshold, fuel damage would also be indicated.

There is no Potential Loss threshold associated with this item.

Plant-Specific

The specified reading is based assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the Containment atmosphere. Because of the very high fuel clad integrity, only small amounts of noble gases would be dissolved in the reactor coolant. The reading is less than that specified for Fuel Clad barrier Loss #2 because no damage to the fuel clad is assumed. Only leakage from the RCS is assumed for this barrier loss threshold.

It is important to recognize that the radiation monitor may be sensitive to shine from the Reactor Vessel or RCS piping. Containment radiation is indicated on 1(2)-RI-5317 A&B. Typical Containment radiation readings at full power are 1 to 1.2 R/hr. The Containment radiation monitors alarm at 6 R/hr and so is operationally significant.

Additionally, high temperatures in Containment may induce a current error in the Mineral Insulated (MI) cable running through Containment to the meter. The CHRRM may not detect this value (6

R/hr) under these conditions. CCNPP replaced the original cable with an improved MI cable under ES199602293 to reduce the induced error to a minimum. When Containment temperature reaches 300 degrees F, the meter will indicate approximately 40 R/hr for a few minutes then drop to approximately 10 R/hr after three hours. This information is to provide guidance on determining the validity of the readings under the specified high temperature conditions. This information has been added as a note.

CCNPP Basis Reference(s):

1. ERPIP-801 Core Damage Assessment Using Containment Radiation Dose Rates

Barrier: Reactor Coolant System

Category: C. Radiation / Coolant Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. Isolation Status

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. Isolation Status

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. Judgment

Degradation Threat: Loss

Threshold:

4. **ANY** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier

Basis:

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

CCNPP Basis Reference(s):

None

Barrier: Reactor Coolant System

Category: E. Judgment

Degradation Threat: Potential Loss

Threshold:

5. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier

Basis:

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

CCNPP Basis Reference(s):

None

Barrier: Containment

Category: A. Core Cooling / Heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: A. Core Cooling / Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CET readings **cannot** be restored < 1,200°F within 15 min.

Basis:

Generic

There is no Loss threshold associated with this item.

The conditions in this threshold represents an imminent core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for Containment failure. In conjunction with the Core Cooling and RCS Leakage criteria in the Fuel Clad and RCS barrier columns, this threshold would result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, there is no "success" path.

The function restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing.

Whether or not the procedures will be effective should be apparent within 15 minutes. The Emergency Director should make the declaration as soon as it is determined that the procedures have been, or will be ineffective.

Plant-Specific

This threshold indicates significant core exit superheating and core uncovery. If Core Exit Thermocouple (CET) readings are greater than 1200°F, Fuel Clad barrier is lost. CETs are a component of the Inadequate Core Cooling Instrumentation system and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. The threshold temperature is consistent with Attachment 3 of ERPIP-802, Core Damage Assessment Using Core Exit Thermocouples. Although clad rupture due to high temperature is not expected for CET readings less than the threshold, temperatures of this magnitude signal significant superheating of the reactor coolant and core uncovery. Events that result in CET readings above the loss threshold are classified severe accidents and lead to a Severe Accident Management Guideline "Badly Damaged (BD)" condition. The BD descriptor signifies possible core overheating to the point that clad ballooning/collapse may occur and portions of the core may have melted.

It must also be assumed the loss of RCS inventory is a result of a loss of the RCS barrier. These conditions, if not mitigated, can lead to core melt which in turn may result in a loss of Containment. Severe accident analyses (e. g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the Reactor Vessel in a significant fraction of the core damage scenarios, and the likelihood of Containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. The phrase "**cannot** be restored <" infers CET readings have exceeded the threshold temperature and procedural guidance used to restore RCS inventory has been attempted but is thus far unsuccessful. Whether or not guidance is effective should be apparent within fifteen minutes. The ED should make the declaration as soon as it is determined the guidance has not been or will not be effective in restoring temperature below the threshold.

CCNPP Basis Reference(s):

1. ERPIP-600 Severe Accident Management
2. ERPIP-601 Severe Accident Management Initial Diagnosis
3. ERPIP-802 Core Damage Assessment Using Core Exit Thermocouples

Barrier: Containment

Degradation Threat: Potential Loss

Category: A. Core Cooling / Heat Removal

Threshold:

2. CET readings > 700°F

   AND

   Reactor vessel water level **cannot** be restored > RVLMS 10 in. alarm within 15 min.

Basis:

Generic

There is no Loss threshold associated with this item.

The conditions in this threshold represents an imminent core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for Containment failure. In conjunction with the Core Cooling and RCS Leakage criteria in the Fuel Clad and RCS barrier columns, this threshold would result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, there is no "success" path.

The function restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing.

Whether or not the procedures will be effective should be apparent within 15 minutes. The Emergency Director should make the declaration as soon as it is determined that the procedures have been, or will be ineffective.

Plant-Specific

This threshold indicates loss of inventory control resulting in significant core exit superheating. The Reactor Vessel Level Monitoring System (RVLMS) can provide indication of potential core uncovery when level decreases to 10 in. alarm. RVLMS is based on the CE Heated Junction Thermocouple (HJTC) system. The HJTC system measures reactor coolant liquid inventory with discrete HJTC sensors located at different levels within a separator tube ranging from the fuel alignment plate (i.e., near top of active fuel) to the Reactor Vessel head. The basic principle of system operation is detection of a temperature difference between heated and unheated thermocouples. Reactor Vessel water level below the top of the core may lead to a Severe

Accident Management Guideline "Badly Damaged (BD)" condition. The BD descriptor signifies possible core overheating to the point that clad ballooning/collapse may occur and portions of the core may have melted. Reactor Vessel water level at the 10 in. alarm signals inadequate coolant inventory, loss of subcooling and the occurrence of possible fuel clad damage.

Core Exit Thermocouples (CETs) are a component of the Inadequate Core Cooling Instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. The RCS Pressure Safety Limit is 2750 psia per CCNPP Technical Specifications. The saturation temperature for this pressure is 682.2°F. Per Action Value Bases Document EOP-24.33, the uncertainty on CET Temperature is +/- 39.8°F. If one or more CETs indicate 722°F (682.2 + 39.8), subcooling has been lost for at least some locations in the core. CET indications at or above 722°F are a clear sign that core heat removal capability is lost or greatly reduced and one fission product barrier, the fuel clad, is threatened due to elevated fuel temperatures. 700°F qualifies as a condition representing a potential loss of the fuel clad barrier (ref. 4).

It must be assumed the loss of RCS inventory is a result of a loss of the RCS barrier. These conditions, if not mitigated, likely lead to core melt which in turn may result in a loss of Containment. Severe accident analyses (e. g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the Reactor Vessel in a significant fraction of the core damage scenarios, and that the likelihood of Containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. The phrase "**cannot** be restored >" infers core uncovery has begun and procedural guidance used to restore RCS inventory has been attempted but is thus far unsuccessful. Whether or not guidance is effective should be apparent within fifteen minutes. The ED should make the declaration as soon as it is determined that the guidance has not been or will not be effective in restoring vessel water level above the threshold.

CCNPP Basis Reference(s):

1. ERPIP-600 Severe Accident Management
2. ERPIP-802 Core Damage Assessment Using Core Exit Thermocouples
3. CEN-152 Emergency Procedure Guidelines
4. EOP-24.33 Action Value Bases Document

Barrier: Containment

Category: B. Inventory

Degradation Threat: Loss

Threshold:

1. A Containment pressure rise followed by a rapid unexplained drop in Containment pressure

Basis:

Generic

Rapid unexplained loss of pressure (i.e., not attributable to Containment spray or condensation effects) following an initial pressure increase from a primary or secondary high energy line break indicates a loss of Containment integrity. Containment pressure should increase as a result of mass and energy release into Containment from a LOCA. Thus, pressure not increasing indicates Containment bypass and a loss of Containment integrity.

This indicator relies on operator recognition of an unexpected response for the condition and therefore does not have a specific value associated with it. The unexpected response is important because it is the indicator for a Containment bypass condition.

Plant-Specific

UFSAR Section 14.20 describes Containment pressure response for a bounding LOCA.

CCNPP Basis Reference(s):

1. UFSAR Section 14.20

Barrier: Containment

Category: B. Inventory

Degradation Threat: Loss

Threshold:

2. Containment pressure or sump level response **not** consistent with LOCA conditions

Basis:

Generic

Containment sump levels should increase as a result of mass and energy release into Containment from a LOCA. Thus, sump level not increasing indicates Containment bypass and a loss of Containment integrity.

This indicator relies on operator recognition of an unexpected response for the condition and therefore does not have a specific value associated with it. The unexpected response is important because it is the indicator for a Containment bypass condition.

Plant-Specific

The Containment pressure and temperature response and Containment sump water temperature response versus time are given in UFSAR Figures 6.2.1-1 through 6.2.1-6b for the most severe LOCAs. During the LOCA injection mode of ECCS operation, Containment sump and RWST levels are monitored to ensure switch-over from injection to cold leg recirculation is initiated automatically and completed via timely operator action.

CCNPP Basis Reference(s):

1. UFSAR Figures 6.2.1-1 through 6.2.1-6b

Barrier: Containment

Category: B. Inventory

Degradation Threat: Loss

Threshold:

3. RUPTURED S/G is also FAULTED outside of Containment

Basis:

Generic

The loss threshold recognizes that SG tube leakage can represent a bypass of the Containment barrier as well as a loss of the RCS barrier.

Users should realize that this threshold and Containment loss B.4 could be considered redundant. This was recognized during the development process. The inclusion of a threshold that uses Emergency Procedure commonly used terms like "ruptured and faulted" adds to the ease of the classification process and has been included based on this human factor concern.

This threshold results in a UE for smaller breaks that; (1) do not exceed the normal charging capacity threshold in RCS leak rate barrier Potential Loss threshold, or (2) do not result in ECCS actuation in RCS SG tube rupture barrier Loss threshold. For larger breaks, RCS barrier threshold criteria would result in an Alert. For SG tube ruptures which may involve multiple steam generators or unisolable secondary line breaks, this threshold would exist in conjunction with RCS barrier thresholds and would result in a Site Area Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.

This threshold addresses the condition in which a ruptured steam generator is also faulted. This condition represents a bypass of the RCS and Containment barriers and is a subset of the Containment loss B.4. In conjunction with RCS leak rate barrier loss threshold, this would always result in the declaration of a Site Area Emergency.

Plant-Specific

A faulted S/G means the existence of secondary side leakage that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized. A ruptured S/G means the existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

Definitions:

Faulted

In a steam generator, the existence of secondary side leakage that results in an uncontrolled drop in steam generator pressure or the steam generator being completely depressurized.

Ruptured

In a steam generator, existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

CCNPP Basis Reference(s):

1. EOP-6 Steam Generator Tube Rupture
2. EOP-8 Functional Recovery Procedure

Barrier: Containment

Category: B. Inventory

Degradation Threat: Loss

Threshold:

4. Primary-to-secondary leakrate > 10 gpm **AND** Unisolable or prolonged steam release from affected S/G to the environment

Basis:

Generic

The loss threshold recognizes that SG tube leakage can represent a bypass of the Containment barrier as well as a loss of the RCS barrier.

Users should realize that the this loss threshold and Containment loss B.3 could be considered redundant. This was recognized during the development process. The inclusion of an threshold that uses Emergency Procedure commonly used terms like "ruptured and faulted" adds to the ease of the classification process and has been included based on this human factor concern.

This threshold results in a UE for smaller breaks that; (1) do not exceed the normal charging capacity threshold in RCS leak rate barrier Potential Loss threshold, or (2) do not result in ECCS actuation in RCS SG tube rupture barrier Loss threshold. For larger breaks, RCS barrier threshold criteria would result in an Alert. For SG tube ruptures which may involve multiple steam generators or unisolable secondary line breaks, this threshold would exist in conjunction with RCS barrier thresholds and would result in a Site Area Emergency. Escalation to General Emergency would be based on "Potential Loss" of the Fuel Clad Barrier.

This threshold addresses SG tube leaks that exceed 10 gpm in conjunction with an unisolable release path to the environment from the affected steam generator. The threshold for establishing the unisolable secondary side release is intended to be a prolonged release of radioactivity from the ruptured steam generator directly to the environment. This could be expected to occur when the main condenser is unavailable to accept the contaminated steam (i.e., SG tube rupture with concurrent loss of off-site power and the ruptured steam generator is required for plant cooldown or a stuck open relief valve). If the main condenser is available, there may be releases via air ejectors, gland seal exhausters, and other similar controlled, and often monitored, pathways. These pathways do not meet the intent of an unisolable release path to the environment. These minor releases are assessed using EALs in Category R.

Plant-Specific

Cool downs conducted to allow controlled isolation of the affected S/G per emergency procedures are not considered prolonged releases. The criterion for prolonged release is met if the objective of EOP-6 or EOP-8 to isolate the affected S/G cannot be met.

Definitions:

Unisolable

A breach or leak that cannot be promptly isolated.

CCNPP Basis Reference(s):

1. EOP-6 Steam Generator Tube Rupture
2. EOP-8 Functional Recovery Procedure

Barrier: Containment

Category: B. Inventory

Degradation Threat: Potential Loss

Threshold:

3. Containment pressure ≥ 50 psig and rising

Basis:

Generic

The site specific pressure is based on the Containment design pressure.

Plant-Specific

This threshold is the Containment design pressure and is in excess of that expected from the design basis loss of coolant accident (LOCA). Proper actuation and operation of the Containment spray system when required should maintain Containment pressure well below the design pressure. The pressure-time responses for the spectrum of LOCAs considered in the plant design basis are described in Section 14 of the UFSAR. The threshold is therefore indicative of a loss of both RCS and Fuel Clad barriers in that it should not be reached without severe core degradation (metal-water reaction) or failure to scram in combination with RCS breach. This condition would be expected to require the declaration of a General Emergency.

CCNPP Basis Reference(s):

1. UFSAR 1.2.5
2. UFSAR 5.1.1
3. UFSAR 14.20

Barrier: Containment

Category: B. Inventory

Degradation Threat: Potential Loss

Threshold:

4. Containment hydrogen concentration ≥ 4%

Basis:

Generic

Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists. The indications of potential loss under this EAL corresponds to some of those leading to Containment potential loss threshold A.1.

Plant-Specific

After a LOCA, the Containment atmosphere is a homogeneous mixture of steam, air, solid and gaseous fission products, hydrogen, and water droplets containing boron and sodium hydroxide. During and following a LOCA, the hydrogen concentration in the Containment results from radiolytic decomposition of water, metal-water reaction, and aluminum/zinc reaction with the spray solution. If hydrogen concentration reaches or exceeds the lower flammability limit (4%) in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside Containment, loss of the Containment barrier could occur. To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers must also have occurred. Since this threshold is also indicative of loss of both Fuel Clad and RCS barriers with the potential loss of the Containment barrier, it therefore will likely warrant declaration of a General Emergency.

CCNPP Basis Reference(s):

1. UFSAR 7.5.8
2. Technical Specifications Table 3.3.10-1
3. OI-41A Hydrogen Recombiners
4. 1C10-ALM ESFAS 14 Alarm Manual, J-09
5. ERPIP-803 Core Damage Assessment Using Hydrogen
6. EOP-8 Functional Recovery Procedure

Barrier: Containment

Category: B. Inventory

Degradation Threat: Potential Loss

Threshold:

5. Containment pressure > 4.25 psig **AND cannot** meet **ANY** of the following conditions:
    - 2 Containment Spray Pumps Operating
    - 3 CACs Operating
    - 1 Containment Spray Pump and 2 CACs Operating

Basis:

Generic

This threshold represents a potential loss of Containment in that the Containment heat removal/depressurization system (e.g., Containment sprays, CACs, etc., but not including Containment venting strategies) are either lost or performing in a degraded manner, as indicated by Containment pressure greater than the setpoint at which the equipment was supposed to have actuated.

Plant-Specific

Two Containment Spray Pumps, three Containment air cooling units or the combination of one Containment spray pump and two Containment air cooling units is defined to be one full train of depressurization equipment. This equipment provides 100% of the required cooling capacity during post-accident conditions. Each Containment spray system consists of a spray pump, spray header, nozzles, valves, piping, instruments, and controls to ensure an operable flow path capable of taking suction from the RWST upon an actuation signal. Each Containment aircooling unit consists of cooling coils, accident backdraft damper, accident fan, service water outlet valves, and controls necessary to ensure an operable service water flow path. The Containment pressure setpoint (4.25 psig) is the pressure at which the equipment should actuate and begin performing its function.

CCNPP Basis Reference(s):

1. 1C08-ALM ESFAS 11 Alarm Manual, G-07, ACTUATION SYSTEM CSAS TRIP
2. Technical Specifications Table 3.3.4-1
3. Technical Specifications 3.6.6
4. EOP-8 Functional Recovery, Appendix 5, CE-3

Barrier: Containment

Category: C. Radiation / Coolant Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: C. Radiation / Coolant Activity

Degradation Threat: Potential Loss

Threshold:

6. Containment radiation monitor (5317A/B) reading > 14,000 R/hr

Basis:

Generic

There is no Loss threshold associated with this item.

The site specific reading is a value which indicates significant fuel damage well in excess of the thresholds associated with both loss of Fuel Clad and loss of RCS barriers. A major release of radioactivity requiring off-site protective actions from core damage is not possible unless a major failure of fuel clad allows radioactive material to be released from the core into the reactor coolant.

Regardless of whether Containment is challenged, this amount of activity in Containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of Containment, such that a General Emergency declaration is warranted.

Plant-Specific

The Containment radiation monitor reading is a value that indicates significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier. NUREG-1228 "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents" states that such readings do not exist when the amount of clad damage is less than 20%. A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure into the reactor coolant has occurred. Regardless of whether the Containment barrier itself is challenged, this amount of activity in Containment could have severe consequences if released. It is, therefore, prudent to treat this as a potential loss of the Containment barrier. The reading is higher than that specified for Fuel Clad barrier Loss #3 and RCS barrier Loss #3. Containment radiation readings at or above the Containment barrier potential loss threshold, therefore, signify a loss of two fission product barriers and potential loss of a third, indicating the need to upgrade the emergency classification to a General Emergency.

Containment radiation is indicated on 1(2)-RI-5317 A&B. Typical Containment radiation readings at full power are 1 to 1.2 R/hr. The Containment radiation monitors alarm at 6 R/hr.

CCNPP Basis Reference(s):

1. ERPIP-801 Core Damage Assessment Using Containment Radiation Dose Rates

Barrier: Containment

Category: D. Isolation Status

Degradation Threat: Loss

Threshold:

5. Failure of **all** valves in **ANY one** line to close

   AND

   Direct downstream pathway to the environment exists after Containment isolation signal

Basis:

Generic

This threshold addresses incomplete Containment isolation that allows direct release to the environment.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in–line charcoal filter does not make a release path indirect since the filter is not effective at removing fission product noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

There is no Potential Loss threshold associated with this item.

Plant-Specific

None

CCNPP Basis Reference(s):

1. EOP-8 Functional Recovery Procedure

Barrier: Containment

Category: D. Isolation Status

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: E. Judgment

Degradation Threat: Loss

Threshold:

6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the Containment barrier

Basis:

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

The Containment barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment barrier status is addressed by Technical Specifications.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

CCNPP Basis Reference(s):

None

Barrier: Containment

Category: E. Judgment

Degradation Threat: Potential Loss

Threshold:

7. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier

Basis:

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

The Containment barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment barrier status is addressed by Technical Specifications.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

CCNPP Basis Reference(s):

None