Enclosure 1 to AEP-NRC-2017-02

AFFIRMATION

I, Q. Shane Lies, being duly sworn, state that I am the Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the U. S. Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

Quinton S. Lies

Q. Shane Lies
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 23 DAY OF May, 2017

Notary Public

My Commission Expires 01/21/2018

Enclosure 2 to AEP-NRC-2017-02

Evaluation of License Amendment Request to Revise Emergency Action Levels

1.0 SUMMARY DESCRIPTION

In accordance with the provisions of 10 CFR 50.90 and 10 CFR 50, Appendix E, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, proposes to amend the CNP Emergency Plan. I&M proposes to update the CNP Emergency Plan by revising the Emergency Action Levels (EALs) used at CNP from the current scheme based on Nuclear Management and Resources Council (NUMARC) and National Environmental Studies Project (NESP) NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels" dated January 1992 (Reference 1), to a scheme based on Nuclear Energy Institute (NEI) 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors" (Reference 2). By Reference 3, the U. S. Nuclear Regulatory Commission (NRC) endorsed NEI 99-01, Revision 6.

As required by 10 CFR 50, Appendix E, Section IV.B.2, a licensee desiring to change its entire EAL scheme shall submit an application for an amendment to its license and receive NRC approval before implementing the change. The proposed EAL scheme would continue to meet the planning standards in 10 CFR 50.47(b) and the requirements of 10 CFR 50, Appendix E.

2.0 DETAILED DESCRIPTION

2.1 Proposed Change

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the CNP Emergency Plan. In 1992, the NRC endorsed Reference 1 as an alternative to NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants (FEMA-REP-1)." In March 2013, the NRC endorsed Reference 2 for use by licensees to upgrade their EALs in accordance with 10 CFR 50, Appendix E. Reference 2 incorporates resolutions to numerous EAL implementation issues, including the NRC EAL frequently-asked questions (FAQs). I&M is proposing to implement an EAL upgrade by using Reference 2 to revise the EALs discussed herein.

The proposed changes involve revising the CNP EAL schemes, which are currently based on Reference 1, to a scheme based on Reference 2. The attached marked-up and clean copies of the EAL Technical Basis Manual (Enclosures 3 and 4 to this letter) provide an explanation and rationale for each CNP EAL, including the necessary plant specific information. Section 1.4 of Reference 2 suggested that the appropriate EALs reflect the availability of enhanced spent fuel pool level instrumentation associated with NRC Order EA-12-051, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" when that instrumentation is available for use. This order has been implemented at CNP and the relevant EALs have been updated to reflect this implementation in the proposed revision.

Enclosure 5 to this letter, NEI 99-01 Revision 6 EAL Comparison Matrix, provides a comparison between the proposed EAL wording, Initiating Conditions (ICs), and Mode Applicability for CNP and the EAL example wording, ICs, and Mode Applicability in Reference 2. This comparison matrix provides a means of assessing CNP differences and deviations from the NRC-endorsed guidance given in Reference 2. Discussion of CNP EAL bases and lists of source document references are given in the EAL Technical Basis Manual (Enclosures 3 and 4).

2.2 Background

I&M previously requested an upgrade to the CNP EALs that used NEI 99-01, Revision 5, as a basis, which was subsequently approved by the NRC. During preparation to implement the changes that were proposed in the new CNP EAL scheme, I&M identified a reactor coolant system (RCS) level instrumentation issue that could not be satisfactorily resolved. As a result, I&M requested withdrawal of the original request for changes to the EAL scheme. The installed instrumentation continues to be acceptable for the existing EAL schemes.

With this request, I&M is proposing an EAL upgrade that revises the CNP EAL schemes, which are currently based on Reference 1, to an EAL scheme based on Reference 2. In order to address the situation leading to the previous withdrawal, I&M is proposing a different EAL for ICs CS1 and CG1, which were affected by the RCS level instrumentation issue. The differences between the proposed CNP EALs and the Reference 2 EALs for NEI ICs CS1 and CG1 are described in Enclosure 5 to this letter, "NEI 99-01 Revision 6 EAL Comparison Matrix."

During the preparation of this proposed EAL scheme, I&M verified that the specific values used as EAL setpoints are within the calibrated range of the referenced instrumentation and that the resolution of the instrumentation is appropriate for the setpoint or indication. The existing instruments in the Radiation Monitoring System are being replaced over the next two years by Mirion instruments. The engineering evaluation associated with this change has verified that the calibrated range of the new instrumentation and the resolution of the instrumentation are appropriate for the EAL setpoint or indication.

3.0 TECHNICAL EVALUATION

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. Reference 2 is the latest guidance endorsed by the NRC that provides guidance to nuclear power plant operators for the development of a site-specific emergency classification scheme. 10 CFR 50.47(b)(4) stipulates that Emergency Plans include a standard emergency classification and action level scheme. This scheme is a fundamental component of an Emergency Plan in that it provides the defined thresholds that will allow site personnel to rapidly implement a range of pre-planned emergency response measures. An emergency classification scheme also facilitates timely decision-making by an Offsite Response Organization concerning the implementation of precautionary or protective actions for the public.

Reference 2 contains ICs, EALs, and fission product barrier status thresholds. It also includes supporting technical basis information, developer notes, and recommended classification instructions for users. The methodology described in this document is consistent with NRC requirements and guidance. In particular, this methodology was endorsed by the NRC in a letter to NEI dated March 28, 2013 (Reference 3), which states that Reference 2 provides an acceptable approach for meeting the requirements of 10 CFR 50.47(b)(4) and the applicable requirements of 10 CFR 50, Appendix E. Reference 3 also reiterated that a licensee desiring to change its entire

EAL scheme shall submit an application for an amendment to its license and receive NRC approval before implementing the change.

The proposed changes to the CNP EAL scheme for adopting the Reference 2 guidance do not reduce the capability to meet the applicable emergency planning requirements established in 10 CFR 50.47 and 10 CFR 50, Appendix E, and will continue to provide consistent emergency classifications. Changes to the CNP Emergency Plan procedures resulting from implementation of the proposed EALs will be evaluated in accordance with the requirements of 10 CFR 50.54(q), subsequent to NRC approval of this amendment request. Accordingly, pursuant to the requirements of 10 CFR 50, Appendix E, Section IV.B.2, I&M requests NRC review and approval of the proposed changes to the EAL schemes in accordance with 10 CFR 50.90.

A Comparison Matrix (Enclosure 5 to this letter) has been developed that provides a tabular format of the ICs, Mode Applicability, and EALs (Threshold Values) in Reference 2, which compares them with the proposed CNP EALs. The matrix provides a comparison of the proposed CNP EALs in terms of differences and deviations from the NRC-endorsed guidance provided in Reference 2. An assessment of the EALs determined whether the proposed EAL wording represents "No Change" from the guidance, a "Difference" from the guidance, or a "Deviation" from the NEI guidance contained in Reference 2. Any items considered to be differences or deviations are based on the definitions provided in RIS 2003-18, "Use of NEI 99-01, Methodology for Development of Emergency Action Levels," and supporting supplements.

The "Difference/Deviation Justification" column in Enclosure 5 identifies each difference between the Reference 2 IC/EAL wording and the CNP IC/EAL wording. An explanation that justifies the reason for each difference is then provided. If any difference has been determined to be a deviation, an explanation is given that states why the deviation from the Reference 2 IC/EAL is acceptable. I&M has identified two differences that represent deviations from the Reference 2 guidance, as shown in Table 3 of Enclosure 5.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.47(b)(4) requires the emergency response plan to meet the following standard:

> A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.

10 CFR 50 Appendix E, Section IV, "Content of Emergency Plans," Item B, "Assessment Actions," states:

2. A licensee desiring to change its entire emergency action level scheme shall submit an application for an amendment to its license and receive NRC approval before implementing the change. Licensees shall follow the change process in 10 CFR 50.54(q) for all other emergency action level changes.

In Reference 3, the NRC states, "Please note that this is considered a significant change to the EAL scheme development methodology and licensees seeking to use this guidance in the development of their EAL scheme must adhere to the requirements of 10 CFR Part 50, Appendix E, Section IV.B.2."

The regulations in 10 CFR 50.54(q) provide direction to licensees seeking to revise their Emergency Plan. The requirements related to nuclear power plant Emergency Plans are given in the standards in 10 CFR 50.47, "Emergency Plans," and the requirements of 10 CFR 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities." 10 CFR 50.47 establishes onsite and offsite emergency response plan standards that are required for the NRC staff to make a positive finding that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. One of these standards, 10 CFR 50.47(b)(4), stipulates that Emergency Plans include a standard emergency classification and action level scheme.

In addition, 10 CFR 50, Appendix E, Section IV.B, "Assessment Actions," stipulates that Emergency Plans include EALs, which are to be used as criteria for determining the need for notification and participation of local and State agencies, and for determining when and what type of protective measures should be considered to protect the health and safety of individuals both onsite and offsite. EALs are to be based on plant conditions and instrumentation, as well as onsite and offsite radiological monitoring.

I&M has evaluated the changes proposed by this LAR and determined that the CNP Emergency Plan will continue to meet applicable regulatory requirements.

4.2 <u>Precedent</u>

This request is similar in nature to requests approved by the NRC for Shearon Harris Nuclear Power Plant (ADAMS Accession No. ML16057A838), H. B. Robinson Steam Electric Plant (ADAMS Accession No. ML16061A472), Comanche Peak Nuclear Power Plant (ADAMS Accession No. ML16137A056), Palo Verde Nuclear Generating Station (ADAMS Accession No. ML16180A109), and Perry Nuclear Power Plant (ADAMS Accession No. ML16158A331).

I&M's request is consistent with the identified precedents and the information presented in this submittal has been developed in consideration of Requests for Additional Information received during the review of the identified precedents.

4.3 <u>No Significant Hazards Consideration Determination</u>

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposes to amend the CNP Emergency Plan Emergency Action Levels (EALs). I&M has evaluated whether a significant hazards consideration is involved with the proposed amendment by addressing the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

Response: No.

The proposed changes to the CNP EALs do not impact the physical function of plant structures, systems, or components (SSC) or the manner in which SSCs perform their design function. EALs are used as criteria for determining the need for notification and participation of local and State agencies, and for determining when and what type of protective measures should be considered within and outside the site boundary to protect health and safety. The proposed changes neither adversely affect accident initiators or precursors, nor alter design assumptions. The proposed changes do not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an initiating event within assumed acceptance limits. No operating procedures or administrative controls that function to prevent or mitigate accidents are affected by the proposed changes.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed changes to the CNP EALs do not involve any physical changes to plant systems or equipment. The proposed changes do not involve the addition of any new equipment. EALs are based on plant conditions, so the proposed changes will not alter the design configuration or the method of plant operation. The proposed changes will not introduce failure modes that could result in a new or different type of accident, and the change does not alter assumptions made in the safety analysis. The proposed changes to the CNP Emergency Plan are not initiators of any accidents.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No

Margin of safety is associated with the ability of the fission product barriers (i.e., fuel cladding, reactor coolant system pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed changes to the CNP EALs do not impact operation of the plant or its response to transients or accidents. The changes do not affect the Technical Specifications or the operating license. The proposed changes do not involve a change in the method of plant operation, and no accident analyses will be affected by the proposed changes.

Additionally, the proposed changes will not relax any criteria used to establish safety limits and will not relax any safety system settings. The safety analysis acceptance criteria are not affected by these changes. The proposed changes will not result in plant operation in a configuration outside the design basis. The proposed changes do not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition. The emergency plan will continue to activate an emergency response commensurate with the extent of degradation of plant safety.

Plant safety margins are established through limiting conditions for operation, limiting safety system settings, and safety limits specified in the technical specifications. The proposed changes involve references to available plant indications to assess conditions for determination of entry into an emergency action level. There is no change to these established safety margins as a result of this change.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusion

In conclusion, based on the considerations discussed above: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the NRC's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

The proposed changes to the EALs maintain the environmental bounds of the current environmental assessment associated with CNP. The proposed changes will not affect plant safety and will not have an adverse effect on the probability of an accident occurring. The proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Nuclear Management and Resources Council (NUMARC) and National Environmental Studies Project (NESP) NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels" dated January 1992.

2. NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," dated November 2012, Agencywide Documents Access and Management System (ADAMS) Accession No. ML12326A805.

3. Letter from M. Thaggard, NRC, to S. Perkins-Grew, NEI, " U.S. Nuclear Regulatory Commission Review and Endorsement of NEI 99-01, Revision 6, Dated November, 2012 (TAC No. D92368)," dated March 28, 2013, ADAMS Accession No. ML12346A463.

Enclosure 3 to AEP-NRC-2017-02

DONALD C. COOK NUCLEAR PLANT EMERGENCY PLAN

EAL TECHNICAL BASIS MANUAL

PAGES MARKED TO SHOW PROPOSED CHANGES (REDLINE)

AEP: D.C. Cook
EAL Technical Basis Manual

(Redline Version)

Revision 0

5/7/17

TABLE OF CONTENTS

1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for D. C. Cook Nuclear Plant (CNP). Decision-makers responsible for implementation of PMP-2080-EPP-101 Emergency Classification, may use this document as a technical reference in support of EAL interpretation. This information may assist the SITE EMERGENCY COORDINATOR (SEC) in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training and for explaining event classifications to off-site officials.

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

Because the information in a basis document can affect emergency classification decision-making (e.g., the Emergency Coordinator refers to it during an event), the NRC staff expects that changes to the basis document will be evaluated in accordance with the provisions of 10 CFR 50.54(q). Additionally, changes to plant AOPs and EOPs that may impact EAL bases shall be evaluated in accordance with the provisions of 10 CFR 50.54(q).

2.0 DISCUSSION

2.1 Background

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

In 1992, the NRC endorsed NUMARC/NESP-007 "Methodology for Development of Emergency Action Levels" as an alternative to NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," EAL guidance.

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

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

Subsequently, Revision 6 of NEI 99-01 "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 (ADAMS Accession Number ML12326A805) (ref. 4.1.1) was issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, CNP conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

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

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

The primary fission product barriers are:

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

B. Reactor Coolant System (RCS): The RCS Barrier 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. Containment Barrier thresholds are used as criteria for escalation of the Emergency Classification Level (ECL) from ALERT to a SITE AREA EMERGENCY or a GENERAL EMERGENCY.

2.3 Fission Product Barrier Classification Criteria

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

Alert:

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

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

2.4 EAL Organization

The CNP EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
  - EALs applicable under any 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 group, assignment of EALs to categories and subcategories:

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

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
R – Abnormal **R**ad Levels / Rad Effluent	1 – Radiological Effluent 2 – Irradiated Fuel Event 3 – Area Radiation Levels
H – **H**azards and Other Conditions Affecting Plant Safety	1 – Security 2 – Seismic Event 3 – Natural or Technological Hazard 4 – Fire 5 – Hazardous Gas 6 – Control Room Evacuation 7 – Emergency Coordinator Judgment
E – ISFSI	1 – Confinement Boundary
Hot Conditions:	
S – **S**ystem Malfunction	1 – Loss of Emergency AC Power 2 – Loss of Vital DC Power 3 – Loss of Control Room Indications 4 – RCS Activity 5 – RCS Leakage 6 – RPS Failure 7 – Loss of Communications 8 – Containment Failure 9 – Hazardous Event Affecting Safety Systems
F – **F**ission Product Barrier Degradation	None
Cold Conditions:	
C – **C**old Shutdown / Refueling System Malfunction	1 – RCS Level 2 – Loss of Emergency AC Power 3 – RCS Temperature 4 – Loss of Vital DC Power 5 – Loss of Communications 6 – Hazardous Event Affecting Safety Systems

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

2.5 Technical Bases Information

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

Category Letter & Title

Subcategory Number & Title

Initiating Condition (IC)

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

EAL Identifier (enclosed in rectangle)

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

1. First character (letter): Corresponds to the EAL category as described above (R, C, H, S, E 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).

Classification (enclosed in rectangle):

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

EAL (enclosed in rectangle)

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

Mode Applicability

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

Definitions:

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

Basis:

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

CNP Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.6 Operating Mode Applicability (ref. 4.1.7)

1 Power Operation

$K_{eff} \geq 0.99$ and reactor thermal power > 5%

2 Startup

$K_{eff} \geq 0.99$ and reactor thermal power ≤ 5%

3 Hot Standby

$K_{eff} < 0.99$ and average coolant temperature ≥ 350°F

4 Hot Shutdown

$K_{eff} < 0.99$ and average coolant temperature 350°F > T_{avg} > 200

5 Cold Shutdown

$K_{eff} < 0.99$ and average coolant temperature ≤ 200°F

6 Refueling

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

D Defueled

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

The mode in effect at the time that an event or condition occurred, and prior to any plant or operator response, is the mode that determines whether or not an IC is applicable. If an event or condition occurs, and results in a mode change before the emergency is declared, the emergency classification level is still based on the mode that existed at the time that the event or condition was initiated (and not when it was declared). Once a different mode is reached, any new event or condition, not related to the original event or condition, requiring emergency classification should be evaluated against the ICs and EALs applicable to the operating mode at the time of the new event or condition. For events that occur in Cold Shutdown or Refueling, escalation is via EALs that are applicable in the Cold Shutdown or Refueling modes, even if Hot Shutdown (or a higher mode) is entered during the subsequent plant response. In particular, the fission product barrier EALs are applicable only to events that initiate in the Hot Shutdown mode or higher.

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Emergency Coordinator must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes, and the informing basis information. In the Recognition Category F matrices, EALs are based on loss or potential loss of Fission Product Barrier Thresholds.

3.1.1 Classification Timeliness

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, "Interim Staff Guidance, Emergency Planning for Nuclear Power Plants" (ref. 4.1.10).

When assessing an EAL that specifies a time duration for the off-normal condition, the "clock" for the EAL time duration runs concurrently with the emergency classification process "clock."

3.1.2 Valid Indications

All emergency classification assessments shall be based upon valid indications, reports or conditions. A valid indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy. For example, verification could be accomplished through an instrument channel check, response on related or redundant indicators, or direct observation by plant personnel. The validation of indications should be completed in a manner that supports timely emergency declaration.

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

3.1.3 Imminent Conditions

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

3.1.4 Planned vs. Unplanned Events

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

3.1.5 Classification Based on Analysis

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

3.1.6 SITE EMERGENCY COORDINATOR (SEC) Judgment

While the EALs have been developed to address a full spectrum of possible events and conditions which may warrant emergency classification, a provision for classification based on operator/management experience and judgment is still necessary. The NEI 99-01 EAL scheme provides the SEC with the ability to classify events and conditions based upon judgment using EALs that are consistent with the Emergency Classification Level (ECL) definitions (refer to Category H). The SEC will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated in the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

3.2 Classification Methodology

To make an emergency classification, the user will compare an event or condition (i.e., the relevant plant indications and reports) to an EAL(s) and determine if the EAL has been met or exceeded. The evaluation of an EAL must be consistent with the related Operating Mode Applicability and Notes. If an EAL has been met or exceeded, the associated IC is likewise met, the emergency classification process "clock" starts, and the ECL must be declared in accordance with plant procedures no later than fifteen minutes after the process "clock" started.

When assessing an EAL that specifies a time duration for the off-normal condition, the "clock" for the EAL time duration runs concurrently with the emergency classification process "clock." For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01 (ref. 4.1.10).

3.2.1 Classification of Multiple Events and Conditions

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

- If an ALERT EAL and a SITE AREA EMERGENCY EAL are met, whether at one unit or at two different units, a SITE AREA EMERGENCY should be declared.

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

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

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

3.2.2 Consideration of Mode Changes During Classification

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

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

3.2.3 Classification of Imminent Conditions

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

3.2.4 Emergency Classification Level Upgrading and Downgrading

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

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

3.2.5 Classification of Short-Lived Events

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

3.2.6 Classification of Transient Conditions

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

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

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

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

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

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

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

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022, Event Report Guidelines 10 CFR 50.72 and 50.73, (ref. 4.1.3) is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR § 50.72 (ref. 4.1.4) within one hour of the discovery of the undeclared event or condition. The licensee should also notify appropriate State and local agencies in accordance with the agreed upon arrangements.

3.2.8 Retraction of an Emergency Declaration

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

4.0 REFERENCES

4.1 Developmental

4.1.1 NEI 99-01 Revision 6, Methodology for the Development of Emergency Action Levels for Non-Passive Reactors, ADAMS Accession Number ML12326A805

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

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

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

4.1.5 10 § CFR 50.73 License Event Report System

4.1.6 CNP UFSAR Figure 1.3-1 Plot Plan

4.1.7 Technical Specifications Table 1.1-1 Modes

4.1.8 PMP-4100-SDR-001 Plant Shutdown Safety and Risk Management

4.1.9 PMP-2010-PRC-001 Procedure Writing

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

4.1.11 CNP Emergency Plan

4.2 Implementing

4.2.1 PMP-2080-EPP-101 Emergency Classification

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

4.2.3 CNP EAL Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

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

ALERT

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

CONTAINMENT CLOSURE

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

As applied to CNP, Containment Closure is established when the requirements of PMP-4100-SDR-001 are met (ref. 4.1.8).

CONFINEMENT BOUNDARY

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As applied to the CNP ISFSI, the CONFINEMENT BOUNDARY is defined to be the Multi-Purpose Canister (MPC).

EMERGENCY ACTION LEVEL

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

EMERGENCY CLASSIFICATION LEVEL

One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are: UNUSUAL EVENT (UE), ALERT, SITE AREA EMERGENCY (SAE) and GENERAL EMERGENCY (GE).

EPA PAGs

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

EXPLOSION

A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

FAULTED

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

FIRE

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

FISSION PRODUCT BARRIER THRESHOLD

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

FLOODING

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

GENERAL EMERGENCY

Events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or hostile actions that result in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

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 CNP 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 CNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTILE FORCE

One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT

The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

IMPEDE(D)

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

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

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

Initiating Condition

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

INTACT (RCS)

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

MAINTAIN

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

~~**Normal Levels**~~

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

OWNER CONTROLLED AREA

The property associated with the station and owned by the company. Access is normally limited to persons entering for official business.

PROJECTILE

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

PROTECTED AREA

The area encompassed by physical barriers to control access to the plant and to the ISFSI. (ref. 4.1.6).

RCS INTACT

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

REDUCED INVENTORY

Operating condition when fuel is in the reactor vessel and Reactor Coolant System level is lower than 3 feet (or more) below the Reactor Vessel flange (ref. 4.1.8).

REFUELING PATHWAY

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

RUPTURED

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

RESTORE

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

SAFETY SYSTEM

A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

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

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

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

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

SECURITY CONDITION

Any security event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A security condition does not involve a hostile action.

SITE AREA EMERGENCY

Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or hostile actions that result 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 Guidelines exposure levels beyond the site boundary.

SITE EMERGENCY COORDINATOR (SEC)

The individual who has the responsibility for event classification, event notification, and approval of protective action recommendations to offsite organizations. It is recognized that during an emergency event this responsibility can be formally turned over from the Shift Manager, to a Site Emergency Coordinator located in the TSC, or to the Emergency Director located in the EOF as the response facilities become activated during an emergency event.

UNISOLABLE

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

UNPLANNED

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

UNUSUAL EVENT

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

VALID

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

VISIBLE DAMAGE

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

5.2 Abbreviations/Acronyms

°F ..... Degrees Fahrenheit
AC ..... Alternating Current
ATWS ..... Anticipated Transient Without Scram
CDE ..... Committed Dose Equivalent
CFR ..... Code of Federal Regulations
CHRM ..... Containment High Range Monitor
CNMT ..... Containment
CNP ..... D. C. Cook Nuclear Plant
CSFST ..... Critical Safety Function Status Tree
D ..... Defueled
DBA ..... Design Basis Accident
DBT ..... Design Basis Threat
DC ..... Direct Current
EAL ..... Emergency Action Level
ECCS ..... Emergency Core Cooling System
ECL ..... Emergency Classification Level
EOF ..... Emergency Operations Facility
EOP ..... Emergency Operating Procedure
EPA ..... Environmental Protection Agency
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
FPB ..... Fission Product Barrier
FSAR ..... Final Safety Analysis Report
GE ..... General Emergency
Hrs ..... Hours
IC ..... Initiating Condition
IPEEE ..... Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI ..... Independent Spent Fuel Storage Installation
K_{eff} ..... Effective Neutron Multiplication Factor
LCO ..... Limiting Condition of Operation
LER ..... Licensee Event Report
LOCA ..... Loss of Coolant Accident
LWR ..... Light Water Reactor
MPC ..... Maximum Permissible Concentration/Multi-Purpose Canister

mR, mRem, mrem, mREM .... milli-Roentgen Equivalent Man
MSL .... Main Steam Line
NEI .... Nuclear Energy Institute
NESP .... National Environmental Studies Project
NPP .... Nuclear Power Plant
NRC .... Nuclear Regulatory Commission
NSSS .... Nuclear Steam Supply System
NORAD .... North American Aerospace Defense Command
(NO)UE .... Notification of Unusual Event
NUMARC .... Nuclear Utility Management and Resource Council
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM .... Off-site Dose Calculation Manual
ORO .... Offsite Response Organization
PA .... Protected Area
PAG .... Protective Action Guideline
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR .... Pressurized Water Reactor
PSIG .... Pounds per Square Inch Gauge
R .... Roentgen
RCC .... Reactor Control Console
RCS .... Reactor Coolant System
Rem, rem, REM .... Roentgen Equivalent Man
RETS .... Radiological Effluent Technical Specifications
RPS .... Reactor Protection System
R(P)V .... Reactor (Pressure) Vessel
RVLIS .... Reactor Vessel Level Indicating System
S/D .... Shutdown
SAR .... Safety Analysis Report
SBO .... Station Blackout
SCBA .... Self-Contained Breathing Apparatus
SEC .... Site Emergency Coordinator
SG .... Steam Generator
SI .... Safety Injection
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
TEDE .... Total Effective Dose Equivalent
TOAF .... Top of Active Fuel
TSC .... Technical Support Center
WOG .... Westinghouse Owners Group

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

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

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

CNP	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
CU1.2	CU1	2
CU2.1	CU2	1
CU3.1	CU3	1
CU3.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1, 2, 3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA3.1	CA3	1, 2
CA6.1	CA6	1
CS1.1	CS1	3
CG1.1	CG1	2
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1, 2 3
HU2.1	HU2	1
HU3.1	HU3	1
HU3.2	HU3	2
HU3.3	HU3	3
HU3.4	HU3	4
HU4.1	HU4	1
HU4.2	HU4	2
HU4.3	HU4	3

CNP	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
HU4.4	HU4	4
HU7.1	HU7	1
HA1.1	HA1	1, 2
HA5.1	HA5	1
HA6.1	HA6	1
HA7.1	HA7	1
HS1.1	HS1	1
HS6.1	HS6	1
HS7.1	HS7	1
HG7.1	HG7	1
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	2
SU5.1	SU4	1, 2, 3
SU6.1	SU5	1
SU6.2	SU5	2
SU7.1	SU6	1, 2, 3
SU8.1	SU7	1, 2
SA1.1	SA1	1
SA3.1	SA2	1
SA6.1	SA5	1
SA9.1	SA9	1
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1

CNP	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
SG1.1	SG1	1
SG2.1	SG8	1
EU1.1	E-HU1	1

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Fission Product Barrier Matrix and Basis

7.3 Safe Operation & Shutdown Rooms/Areas Tables R-2 & H-2 Bases

Category R – Abnormal Rad Release / Rad Effluent

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

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

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

Events of this category pertain to the following subcategories:

1. Radiological Effluent

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

2. Irradiated Fuel Event

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

3. Area Radiation Levels

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

ATTACHMENT 1
EAL Bases

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.1 Unusual Event

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

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

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

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

Table R-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	ALERT	UE
Gaseous	Unit Vent Noble Gas	VRS-1500 (2500)	3.3E+00 μCi/cc	3.3E-01 μCi/cc	3.3E-02 μCi/cc	4.2E-03 μCi/cc
Gaseous	Gland Seal Leakoff	SRA-1800 (2800)	1.6E+02 μCi/cc	1.6E+01 μCi/cc	1.6E+00 μCi/cc	1.4E-01 μCi/cc
Gaseous	Steam Jet Air Ejector	SRA-1900 (2900)	1.5E+04 μCi/cc	1.5E+03 μCi/cc	1.5E+02 μCi/cc	1.3E+01 μCi/cc
Liquid	Radwaste Effluent	RRS-1001	----	----	----	4.6E+04 cpm
Liquid	SG Blowdown	R-19	----	----	----	1.7E+03 cpm
Liquid	SG Blowdown	DRS-3100/4100	----	----	----	1.2E+04 cpm
Liquid	SG Blowdown Treatment	R-24	----	----	----	2.9E+04 cpm
Liquid	SG Blowdown Treatment	DRS-3200/4200	----	----	----	1.2E+05 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

The column "UE" gaseous and liquid release values in Table R-1 represent two times the appropriate ODCM release rate limits associated with the specified monitors (ref. 1, 2).

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

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

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

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

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

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

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

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

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

CNP Basis Reference(s):

1. EP-CALC-CNP-1601, Radiological Effluent EAL Threshold Values
2. PMP-6010-OSD-001, Off Site Dose Calculation Manual
3. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

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

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

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

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

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

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

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

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

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

CNP Basis Reference(s):

1. PMP-6010-OSD-001, Off Site Dose Calculation Manual
2. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.1 Alert

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

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

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

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

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

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	ALERT	UE
Gaseous	Unit Vent Noble Gas	VRS-1500 (2500)	3.3E+00 µCi/cc	3.3E-01 µCi/cc	3.3E-02 µCi/cc	4.2E-03 µCi/cc
Gaseous	Gland Seal Leakoff	SRA-1800 (2800)	1.6E+02 µCi/cc	1.6E+01 µCi/cc	1.6E+00 µCi/cc	1.4E-01 µCi/cc
Gaseous	Steam Jet Air Ejector	SRA-1900 (2900)	1.5E+04 µCi/cc	1.5E+03 µCi/cc	1.5E+02 µCi/cc	1.3E+01 µCi/cc
Liquid	Radwaste Effluent	RRS-1001	----	----	----	4.6E+04 cpm
Liquid	SG Blowdown	R-19	----	----	----	1.7E+03 cpm
Liquid	SG Blowdown	DRS-3100/4100	----	----	----	1.2E+04 cpm
Liquid	SG Blowdown Treatment	R-24	----	----	----	2.9E+04 cpm
Liquid	SG Blowdown Treatment	DRS-3200/4200	----	----	----	1.2E+05 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either:

- 10 mrem TEDE
- 50 mrem CDE Thyroid

The column "ALERT" gaseous effluent release values in Table R-1 correspond to calculated doses of 1% (10% of the SAE thresholds) of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

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

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

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

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

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

CNP Basis Reference(s):

1. EP-CALC-CNP-1601, Radiological Effluent EAL Threshold Values
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

Dose assessments are performed by computer-based or manual methods (ref. 1).

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

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

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

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

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

CNP Basis Reference(s):

1. PMP-2080-EPP-108 Initial Dose Assessment
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.3 Alert

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

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

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

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

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

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

CNP Basis Reference(s):

1. PMP-6010-OSD-001, Off Site Dose Calculation Manual
2. NEI 99-01 AA1

ATTACHMENT 1
EAL Bases

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.4 Alert

Field survey results indicate **EITHER** of the following at or beyond the site boundary:

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

RMT-2080-EOF-001, Activation and Operation of the EOF provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

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

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

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

~~Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have~~

~~stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.~~

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

CNP Basis Reference(s):

1. RMT-2080-EOF-001 Activation and Operation of the EOF
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.1 Site Area Emergency

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

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

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

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

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

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	ALERT	UE
Gaseous	Unit Vent Noble Gas	VRS-1500 (2500)	3.3E+00 μCi/cc	3.3E-01 μCi/cc	3.3E-02 μCi/cc	4.2E-03 μCi/cc
Gaseous	Gland Seal Leakoff	SRA-1800 (2800)	1.6E+02 μCi/cc	1.6E+01 μCi/cc	1.6E+00 μCi/cc	1.4E-01 μCi/cc
Gaseous	Steam Jet Air Ejector	SRA-1900 (2900)	1.5E+04 μCi/cc	1.5E+03 μCi/cc	1.5E+02 μCi/cc	1.3E+01 μCi/cc
Liquid	Radwaste Effluent	RRS-1001	----	----	----	4.6E+04 cpm
Liquid	SG Blowdown	R-19	----	----	----	1.7E+03 cpm
Liquid	SG Blowdown	DRS-3100/4100	----	----	----	1.2E+04 cpm
Liquid	SG Blowdown Treatment	R-24	----	----	----	2.9E+04 cpm
Liquid	SG Blowdown Treatment	DRS-3200/4200	----	----	----	1.2E+05 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either:

- 100 mrem TEDE
- 500 mrem CDE Thyroid

The column "SAE" gaseous effluent release value in Table R-1 corresponds to calculated doses of 10% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

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

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

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

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

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

CNP Basis Reference(s):

1. EP-CALC-CNP-1601-CNP-1601, Radiological Effluent EAL Threshold Values
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.2	**Site Area Emergency**
Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the site boundary (Note 4)	

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

Mode Applicability:

All

Definition(s):

None

Basis:

Dose assessments are performed by computer-based and manual methods (ref. 1).

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

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

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

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

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

CNP Basis Reference(s):

1. PMP-2080-EPP-108 Initial Dose Assessment
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.3 Site Area Emergency

Field survey results indicate **EITHER** of the following at or beyond the site boundary:

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

RMT-2080-EOF-001, Activation and Operation of the EOF provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

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

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

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

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

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

CNP Basis Reference(s):

1. RMT-2080-EOF-001 Activation and Operation of the EOF
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.1 General Emergency

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

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

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

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

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

Table R-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	ALERT	UE
Gaseous	Unit Vent Noble Gas	VRS-1500 (2500)	3.3E+00 µCi/cc	3.3E-01 µCi/cc	3.3E-02 µCi/cc	4.2E-03 µCi/cc
Gaseous	Gland Seal Leakoff	SRA-1800 (2800)	1.6E+02 µCi/cc	1.6E+01 µCi/cc	1.6E+00 µCi/cc	1.4E-01 µCi/cc
Gaseous	Steam Jet Air Ejector	SRA-1900 (2900)	1.5E+04 µCi/cc	1.5E+03 µCi/cc	1.5E+02 µCi/cc	1.3E+01 µCi/cc
Liquid	Radwaste Effluent	RRS-1001	----	----	----	4.6E+04 cpm
Liquid	SG Blowdown	R-19	----	----	----	1.7E+03 cpm
Liquid	SG Blowdown	DRS-3100/4100	----	----	----	1.2E+04 cpm
Liquid	SG Blowdown Treatment	R-24	----	----	----	2.9E+04 cpm
Liquid	SG Blowdown Treatment	DRS-3200/4200	----	----	----	1.2E+05 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either:

- 1000 mrem TEDE
- 5000 mrem CDE Thyroid

The column "GE" gaseous effluent release values in Table R-1 correspond to calculated doses of 100% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

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

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

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

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

CNP Basis Reference(s):

1. EP-CALC-CNP-1601, Radiological Effluent EAL Threshold Values
2. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

Dose assessments are performed by computer-based and manual methods (ref. 1).

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

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

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

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

CNP Basis Reference(s):

1. PMP-2080-EPP-108 Initial Dose Assessment
2. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.3 General Emergency

Field survey results indicate **EITHER** of the following at or beyond the site boundary:

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

RMT-2080-EOF-001, Activation and Operation of the EOF provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

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

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

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

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

CNP Basis Reference(s):

ATTACHMENT 1
EAL Bases

1. RMT-2080-EOF-001 Activation and Operation of the EOF
2. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Unplanned loss of water level above irradiated fuel

EAL:

RU2.1 Unusual Event

UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication

AND

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

- VRS-1101/1201, Unit 1 Upper Containment
- VRS-2101/2201, Unit 2 Upper Containment
- R-5 Spent Fuel Area
- VRS-5006 Spent Fuel Area

Mode Applicability:

All

Definition(s):

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

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

Basis:

The low water level alarm in this EAL refers to the Spent Fuel Pool (SFP) low (Panel 134 Drop 2) or low-low level alarms (Panel 105 Drop 27 or Panel 205 Drop 27) (ref. 1). During the fuel transfer phase of refueling operations, the fuel transfer canal is normally in communication with the spent fuel pool and the refueling cavity in the Containment is in communication with the fuel transfer canal when the fuel transfer tube is open. A lowering in water level in the SFP, fuel transfer canal or refueling cavity is therefore sensed by the SFP low level alarm. Neither the refueling cavity nor the fuel transfer canal is equipped with a low level alarm (ref. 1).

Technical Specification Section 3.7.14 (ref. 5) requires at least 23 ft of water above the SFP storage racks. Technical Specification Section 3.9.6 (ref. 4) requires at least 23 ft of water above the Reactor Vessel flange in the refueling cavity. During refueling, this maintains sufficient water level in the fuel transfer canal, refueling cavity, and SFP to retain iodine fission product activity in the water in the event of a fuel handling accident.

The listed radiation monitors are those expected to see increase area radiation levels as a result of a loss of REFUELING PATHWAY inventory (ref. 1, 2, 3). Increasing radiation indications on these monitors in the absence of indications of decreasing REFUELING PATHWAY level are not classifiable under this EAL.

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

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

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

The effects of planned evolutions should be considered. For example, a refueling bridge area radiation monitor reading may increase due to planned evolutions such as lifting of the reactor vessel head or movement of a fuel assembly. Note that this EAL is applicable only in cases where the elevated reading is due to an unplanned loss of water level.

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

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

CNP Basis Reference(s):

1. 12-OHP-0422-018-002 Loss of Refueling Water Level During Refueling Operations – Local Actions
2. 12-OHP-4022-018-003 Irradiated Fuel Handling Accident in Containment Building – Local Actions
3. 12-OHP-4022-018-004 Irradiated Fuel Handling Accident in Containment Building – Control Room Actions
4. Technical Specification Section 3.9.6 Refueling Cavity Water Level
5. Technical Specification Section 3.7.14 Fuel Storage Pool Water Level
6. NEI 99-01 AU2

ATTACHMENT 1
EAL Bases

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

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

Mode Applicability:

All

Definition(s):

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

Basis:

This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool ~~(see Developer Notes)~~. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant. This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with ~~IC E-H~~U1.1.

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

~~EAL #1~~

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

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

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

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

~~EAL #3~~

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

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

CNP Basis Reference(s):

1. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by High alarm on **any** of the following radiation monitors:

- VRS-1101/1201, Unit 1 Upper Containment
- VRS-2101/2201, Unit 2 Upper Containment
- R-5 Spent Fuel Area
- VRS-5006 Spent Fuel Area

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with ~~IC~~ E~~-H~~U1.1.

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

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

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

~~readings should be considered in combination with other available indications of inventory loss.~~

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

This EAL addresses a release of radioactive material caused by mechanical damage to irradiated fuel. Damaging events may include the dropping, bumping or binding of an assembly, or dropping a heavy load onto an assembly. A rise in readings on radiation monitors should be considered in conjunction with in-plant reports or observations of a potential fuel damaging event (e.g., a fuel handling accident).~~EAL #3Spent fuel pool water level at this value is within the lower end of the level range necessary to prevent significant dose consequences from direct gamma radiation to personnel performing operations in the vicinity of the spent fuel pool. This condition reflects a significant loss of spent fuel pool water inventory and thus it is also a precursor to a loss of the ability to adequately cool the irradiated fuel assembles stored in the pool.~~

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

CNP Basis Reference(s):

1. 12-OHP-4022-018-003 Irradiated Fuel Handling Accident in Containment Building – Local Actions
2. 12-OHP-4022-018-004 Irradiated Fuel Handling Accident in Containment Building – Control Room Actions
3. 12-OHP-4022-018-005 Irradiated Fuel Handling Accident in Spent Fuel Storage Area – Local Actions
4. 12-OHP-4022-018-006 Irradiated Fuel Handling Accident in Spent Fuel Storage Area – Control Room Actions
5. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.3 Alert

Lowering of spent fuel pool level to 9 ft. 6 in. on 1(2)-RLI-502-CRI Spent Fuel Pit Level Indication (8 ft. 10 in. on local ruler)

Mode Applicability:

All

Definition(s):

None

Basis:

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

For CNP SFP Level 2 is plant elevation 630 ft. 10.5 in. or 9 ft. 6 in. as indicated on 1(2)-RLI-502-CRI in the Control Room or 1(2)-RLI-502-BATT back-up indicator (ref. 2). This level corresponds to 8 ft. 10 in. on the SFP ruler.

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

~~This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC E-HU1.~~

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

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

~~A drop in water level above irradiated fuel within the reactor vessel may be classified in~~

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

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

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

Escalation of the emergency classification level would be via IC~~s~~ ~~AS1~~ RS1or ~~AS2~~ RS2~~(*see AS2 Developer Notes*)~~.

CNP Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. EC-0000052892 Spent Fuel Pool Level for NRC Order EA-12-051
3. NEI 99-01 AA2

ATTACHMENT 1
EAL Bases

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RS2.1 Site Area Emergency

Lowering of spent fuel pool level to 0 ft. on 1(2)-RLI-502-CRI Spent Fuel Pit Level Indication

Mode Applicability:

All

Definition(s):

None

Basis:

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

For CNP SFP Level 3 is plant elevation 620 ft. 10.5 in. However, the SFP level instrument lower range (0 ft.) corresponds to plant elevation 621 ft. 6 in. Therefore an indicated level of 0 ft. on 1(2)-RLI-502-CRI in the Control Room or 1(2)-RLI-502-BATT back-up indicator is used as indicated Level 3 (ref. 2).

This ~~IC~~ EAL addresses a significant loss of spent fuel pool inventory control and makeup capability leading to IMMINENT fuel damage. This condition entails major failures of plant functions needed for protection of the public and thus warrant a SITE AREA EMERGENCY declaration.

It is recognized that this IC would likely not be met until well after another SITE AREA EMERGENCY IC was met; however, it is included to provide classification diversity.

Escalation of the emergency classification level would be via IC ~~AG1~~ RG1 or ~~AG2~~RG2.

CNP Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. EC-0000052892 Spent Fuel Pool Level for NRC Order EA-12-051
3. NEI 99-01 AS2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RG2.1	**General Emergency**
Spent fuel pool level **cannot** be restored to at least 0 ft. on 1(2)-RLI-502-CRI Spent Fuel Pit Level Indication for ≥ 60 min. (Note 1)	

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

Mode Applicability:

All

Definition(s):

None

Basis:

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

For CNP SFP Level 3 is plant elevation 620 ft. 10.5 in. However, the SFP level instrument lower range (0 ft.) corresponds to plant elevation 621 ft. 6 in. Therefore an indicated level of 0 ft. on 1(2)-RLI-502-CRI in the Control Room or 1(2)-RLI-502-BATT back-up indicator is used as indicated Level 3 (ref. 2).

This ~~IC~~ EAL addresses a significant loss of spent fuel pool inventory control and makeup capability leading to a prolonged uncovery of spent fuel. This condition will lead to fuel damage and a radiological release to the environment.

It is recognized that this IC would likely not be met until well after another GENERAL EMERGENCY IC was met; however, it is included to provide classification diversity.

CNP Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. EC-0000052892 Spent Fuel Pool Level for NRC Order EA-12-051
3. NEI 99-01 AG2

ATTACHMENT 1
EAL Bases

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

RA3.1 Alert

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

- Unit 1 Control Room (ERS-7401)
- Unit 2 Control Room (ERS-8401)
- Central Alarm Station (by survey)
- Secondary Alarm Station (by survey)

Mode Applicability:

All

Definition(s):

None

Basis:

Areas that meet this threshold include the Control Rooms and the Central Alarm Station (CAS) and Secondary Alarm Station (SAS). ERS-7401 (ERS-8401) monitor the Control Rooms for area radiation (ref. 1). The CAS and SAS are included in this EAL because of their importance to permitting access to areas required to assure safe plant operations (ref. 1).

There is no permanently installed CAS or SAS area radiation monitors that may be used to assess this EAL threshold. Therefore this threshold must be assessed via local radiation survey for the CAS and SAS (ref. 1).

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

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

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

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

CNP Basis Reference(s):

1. FSAR Table 11.3-1 Radiation Monitoring System Channel Sensitivities and Detecting Medium
2. NEI 99-01 AA3

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

RA3.2 Alert

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

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

Table R-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Auxiliary Building 573'	4, 5
Auxiliary Building 587' (including D/G room)	1, 2, 3, 4, 5
Auxiliary Building 591'	1, 2, 3, 4
Auxiliary Building 609' (including 4kV room)	1, 2, 3, 4, 5
Auxiliary Building 633'	1, 2, 3, 4
Turbine Building (**All** Levels)	1, 2, 3
Turbine Building 591'	4, 5
Screenhouse	1, 2, 3, 4, 5

Mode Applicability:

All

Definition(s):

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

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

Basis:

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

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

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

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

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

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

CNP Basis Reference(s):

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

Category E – Independent Spent Fuel Storage Installation (ISFSI)

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

An independent spent fuel storage installation (ISFSI) is a complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage. A significant amount of the radioactive material contained within a canister must escape its packaging and enter the biosphere for there to be a significant environmental effect resulting from an accident involving the dry storage of spent nuclear fuel.

An UNUSUAL EVENT is declared on the basis of the occurrence of an event of sufficient magnitude that a loaded cask confinement boundary is damaged or violated.

Category: ISFSI

Subcategory: Confinement Boundary

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Unusual Event

Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading:

- > 60 mrem/hr (gamma + neutron) on the top of the overpack
- > 600 mrem/hr gamma + neutron) on the side of the overpack excluding inlet and outlet ducts

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY-. The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As applied to the CNP ISFSI, the CONFINEMENT BOUNDARY is defined to be the Multi-Purpose Canister (MPC).

Basis:

Overpacks are the HI-STORM 100 casks which receive and contain the sealed MPCs for interim storage in the ISFSI. They provide gamma and neutron shielding, and provide for ventilated air flow to promote heat transfer from the MPC to the environs. The term overpack does not include the transfer cask (ref. 1).

The value shown represents 2 times the maximum overpack surface dose rates specified in Section 5.7 of the ISFSI Certificate of Compliance Technical Specifications for radiation external to a loaded MPC overpack (ref. 1).

This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage.

The existence of "damage" is determined by radiological survey. The technical specification multiple of "2 times", which is also used in Recognition Category ~~A~~R IC R~~A~~U1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

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

CNP Basis Reference(s):

1. Certificate of Compliance No. 1014 Holtec International HI-STORM 100 Cask System Safety Evaluation Report Amendment 1 Appendix A Technical Specifications Section 5.7
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 - Refueling, D – Defueled).

The events of this category pertain to the following subcategories:

1. RCS Level

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

2. Loss of Emergency AC Power

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

3. RCS Temperature

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

4. Loss of Vital DC Power

Loss of emergency 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 vital plant 250 VDC power sources.

5. Loss of Communications

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

6. Hazardous Event Affecting Safety Systems

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

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory

EAL:

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

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

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

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

Basis:

With the plant in Cold Shutdown, RCS water level is normally established by 1 (2)-OHP-4021-002-005, RCS Draining (ref. 1). If RCS level is being controlled below the pressurizer low level setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

With the plant in Refueling mode, RCS water level is normally maintained at or above the reactor vessel flange (Technical Specification LCO 3.9.6 requires at least 23 ft. of water above the top of the reactor vessel flange in the refueling cavity during refueling operations) (ref. 2).

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

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

This EAL ~~#1~~ recognizes that the minimum required ~~(reactor vessel/~~RCS ~~[*PWR*] or RPV [*BWR*])~~ level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.

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

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

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

CNP Basis Reference(s):

1. 1(2)-OHP-4021-002-005, RCS Draining
2. Technical Specification Section 3.9.6 Refueling Cavity Water Level
3. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory

EAL:

CU1.2 Unusual Event

RCS water level cannot be monitored

AND EITHER

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

Table C-1 Sumps / Tanks
• Containment Sumps • Auxiliary Building Sumps • RWST • RCDT

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

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

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

Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available.

In this EAL, all water level indication is unavailable and the RCS inventory loss must be detected by indirect leakage indications. 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. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor ~~(reactor vessel/~~RCS ~~[PWR]~~

~~or RPV [BWR])~~ level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

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

~~EAL #1 recognizes that the minimum required (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.~~

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

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

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

CNP Basis Reference(s):

1. 1(2)-OHP-4022-002-020 Excessive Reactor Coolant Leakage
2. 1(2)-OHP-4021-002-005, RCS Draining
3. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Significant loss of RCS inventory

EAL:

CA1.1	**Alert**
Loss of RCS inventory as indicated by RCS level < 614.0 ft.	

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

None

Basis:

614.0 ft. corresponds to midloop and is the minimum allowed RCS level for operation of RHR (ref.1)

RCS level cannot be measured below 612 feet on NLI-1000, High Resolution – RCS Full Range Level Indication, which is below the bottom ID of the hot leg inlet. Should RCS level drop below this point it is assumed water level cannot be monitored other than visually.

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

For this EAL ~~#1~~, a lowering of RCS water level below 614.0 ft. ~~(site-specific level) ft.~~ indicates that operator actions have not been successful in restoring and maintaining RCS ~~(reactor vessel/RCS [PWR] or RPV [BWR])~~ water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

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

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

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

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

CNP Basis Reference(s):

1. 1(2)-OHP-4022-017-001 Loss of RHR Cooling
2. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Significant loss of RCS inventory

EAL:

CA1.2 Alert

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

AND EITHER

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

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

Table C-1 Sumps / Tanks
• Containment Sumps • Auxiliary Building Sumps • RWST • RCDT

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

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

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

Basis:

<u>In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available.</u>

<u>In the Refuel mode, the RCS is not intact and RPV level may be monitored by different means, including the ability to monitor level visually.</u>

<u>In this EAL, all RCS water level indication would be unavailable for greater than 15 minutes, and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). Surveillance procedures provide instructions for calculating primary system leak rate by manual or computer-based water inventory balances. 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. If the make-up rate to the RCS</u>

unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

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

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

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

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

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

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

CNP Basis Reference(s):

1. 1(2)-OHP-4022-002-020 Excessive Reactor Coolant Leakage
2. 1(2)-OHP-4021-002-005, RCS Draining
3. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.1 Site Area Emergency

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

AND

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

- UNPLANNED increase in **any** Table C-1 sump/tank level of sufficient magnitude to indicate core uncovery
- High alarm on Containment radiation monitor VRA-1310 (2310) or VRA-1410(2410)
- Erratic Source Range Monitor indication

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

Table C-1 Sumps / Tanks
• Containment Sumps • Auxiliary Building Sumps • RWST • RCDT

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

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

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

Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available.

In the Refueling mode, the RCS is not intact and reactor vessel level may be monitored by different means, including the ability to monitor level visually.

In this EAL, all RCS water level indication would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). Surveillance procedures provide instructions for calculating primary system leak rate by manual or computer-based water inventory balances. 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. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

The reactor vessel inventory loss may be detected by the containment radiation monitors VRA-1310 (2310) or 1410 (2410) or erratic Source Range Monitor indication. 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 a high alarm on containment high range radiation monitors (ref. 3).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations (ref. 4).

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

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

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

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

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

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

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

CNP Basis Reference(s):

1. 1(2)-OHP-4022-002-020 Excessive Reactor Coolant Leakage
2. 1(2)-OHP-4021-002-005, RCS Draining
3. Calculation No. 1-2-UNC-421 Post Accident High Range Containment Area Radiation Monitoring Loop Uncertainty Calculation
4. Nuclear Safety Analysis Center (NSAC), 1980, "Analysis of Three Mile Island - Unit 2 Accident," NSAC-1
5. NEI 99-01 CS1

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CG1.1 General Emergency

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

AND

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

- UNPLANNED increase in **any** Table C-1 sump/tank level of sufficient magnitude to indicate core uncovery
- High alarm on Containment radiation monitor VRA-1310 (2310) or VRA-1410(2410)
- Erratic Source Range Monitor indication

AND

Any Containment Challenge indication, Table C-2

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

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

Table C-1 Sumps / Tanks
• Containment Sumps • Auxiliary Building Sumps • RWST • RCDT

Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • Containment hydrogen concentration ≥ 4% • Unplanned rise in Containment pressure

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

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

As applied to CNP, Containment Closure is established when the requirements of PMP-4100-SDR-001 are met.

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

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

Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available.

In the Refueling mode, the RCS is not intact and RPV level may be monitored by different means, including the ability to monitor level visually.

In this EAL, all RCS water level indication would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). Surveillance procedures provide instructions for calculating primary system leak rate by manual or computer-based water inventory balances. 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. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

The reactor vessel inventory loss may be detected by the containment radiation monitors VRA-1310 (2310) or 1410 (2410) or erratic Source Range Monitor indication. 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 a high alarm on containment high range radiation monitors (ref. 3).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations (ref. 4).

Three conditions are associated with a challenge to containment integrity:

1. CONTAINMENT CLOSURE not established - The status of Containment closure is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 5). If containment closure is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to GE would not occur.
2. Containment hydrogen ≥ 4% - The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations. CNP is equipped with a Post-Accident Containment Hydrogen Monitoring System (PACHMS) that is capable of continuously measuring the concentration of hydrogen in the containment atmosphere

following a significant beyond design-basis accident for accident mitigation, including emergency planning. PACHMS is comprised of two sampling-analyzing-control trains. Each train has two subsystems - the hydrogen analyzer panels and the remote control panels (ref. 6).

3. UNPLANNED rise in Containment pressure - An unplanned pressure rise in containment while in cold Shutdown or Refueling modes can threaten Containment Closure capability and thus containment potentially cannot be relied upon as a barrier to fission product release.

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

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

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

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

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

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

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

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

CNP Basis Reference(s):

1. 1(2)-OHP-4022-002-020 Excessive Reactor Coolant Leakage
2. 1(2)-OHP-4021-002-005, RCS Draining
3. Calculation No. 1-2-UNC-421 Post Accident High Range Containment Area Radiation Monitoring Loop Uncertainty Calculation
4. Nuclear Safety Analysis Center (NSAC), 1980, "Analysis of Three Mile Island - Unit 2 Accident," NSAC-1
5. PMP-4100-SDR-001 Plant Shutdown Safety and Risk Management
6. UFSAR Section 7.8.2 Post-Accident Containment Hydrogen Monitoring
7. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CU2.1 Unusual Event

AC power capability, Table C-3, to emergency 4.16 kV buses T11A (T21A) and T11D (T21D) reduced to a single power source for ≥ 15 min. (Note 1)

AND

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

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

Table C-3 AC Power Sources
Offsite: • Reserve Auxiliary Xmr TR101AB (TR201AB) • Reserve Auxiliary Xmr TR101CD (TR201CD) • 69/4.16 kV Alternate Xmr TR12EP-1 • Main Xmr TR1 (TR2) backfeed (only if already aligned) **Onsite:** • EDG 1AB (2AB) • EDG 1CD (2CD)

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling, D - Defueled

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

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

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

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

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

Basis:

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

The condition indicated by this EAL is the degradation of the offsite and onsite power sources such that any additional single failure would result in a loss of all AC power to the emergency buses.

A list of onsite and offsite AC power sources credited for this EAL are specified in Table C-3.

4.16KV buses T11A (T21A) and T11D (T21D) are the emergency (essential) buses (ref. 1). While generating, auxiliary power is normally supplied from the generator terminals through the unit auxiliary transformers (TR1AB and TR1CD for Unit 1 and TR2AB and TR2CD for Unit 2). When the plant trips or the plant is shutdown the station auxiliaries are transferred to the preferred offsite power source (that is, to reserve auxiliary transformers TR101AB and TR101CD for Unit 1 and TR201AB and TR201CD for Unit 2) to assure continued power to equipment when the main generator is off-line (ref. 1).

In addition, an alternate offsite power source, a 69/4.16kV transformer (TR12EP-1), located at the plant site, has the necessary capacity to operate one train of the engineered safeguard equipment in one unit while supplying one train of the safe shutdown power in the other.

T11A (T21A) and T11D (T21D) also each have an emergency diesel generator which supply onsite electrical power to the bus automatically in the event that the preferred offsite sources become unavailable (ref. 1).

Another method to obtain offsite power is by backfeeding the emergency buses through the main transformer and unit auxiliary transformers. This is only done during cold shutdown when no other power sources are available (ref. 1, 3). Credit is only taken for this source if already aligned as it requires removal of the main generator disconnect links.

The Supplemental Diesel Generators (SDGs) are not credited as an AC power source for this EAL.

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

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

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

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

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel

generators) with a single train of emergency buses being back-fed from the unit main generator.

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

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

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

CNP Basis Reference(s):

1. UFSAR Figure 8.1-1A(B) Main Auxiliary One-Line Diagram
2. UFSAR Section 8.0 Electrical Systems
3. 1(2)-OHP-4022-001-005 Loss of Offsite Power with Reactor Shutdown
4. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CA2.1	**Alert**
Loss of **all** offsite and **all** onsite AC power to emergency 4.16KV buses T11A (T21A) and T11D (T21D) for ≥ 15 min. (Note 1)	

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

Mode Applicability:

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

Basis:

4.16KV buses T11A (T21A) and T11D (T21D) are the emergency (essential) buses (ref. 1). While generating, auxiliary power is normally supplied from the generator terminals through the unit auxiliary transformers (TR1AB and TR1CD for Unit 1 and TR2AB and TR2CD for Unit 2). When the plant trips or the plant is shutdown the station auxiliaries are transferred to the preferred offsite power source (that is, to reserve auxiliary transformers TR101AB and TR101CD for Unit 1 and TR201AB and TR201CD for Unit 2) to assure continued power to equipment when the main generator is off-line (ref. 1).

In addition, an alternate offsite power source, a 69/4.16kV transformer (TR12EP-1), located at the plant site, has the necessary capacity to operate one train of the engineered safeguard equipment in one unit while supplying one train of the safe shutdown power in the other.

T11A (T21A) and T11D (T21D) also each have an emergency diesel generator which supply electrical power to the bus automatically in the event that the preferred offsite sources become unavailable (ref. 1).

Another method to obtain offsite power is by backfeeding the emergency buses through the main transformer and unit auxiliary transformers. This is only done during cold shutdown when no other power sources are available (ref. 1, 3). Credit is only taken for this source if already aligned as it requires removal of the main generator disconnect links.

The Supplemental Diesel Generators (SDGs) or any other alternative AC power source capable of powering an emergency bus can also be credited as an AC power source for this EAL.

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

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

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

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

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

CNP Basis Reference(s):

1. UFSAR Figure 8.1-1A(B) Main Auxiliary One-Line Diagram
2. UFSAR Section 8.0 Electrical Systems
3. 1(2)-OHP-4022-001-005 Loss of Offsite Power with Reactor Shutdown
4. NEI 99-01 CA2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.1	**Unusual Event**
UNPLANNED increase in RCS temperature to > 200°F (Note 10)	

Note 10: Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss of decay heat removal.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

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

Basis:

The following instrumentation is capable of providing indication of an RCS temperature rise that approaches the Technical Specification cold shutdown temperature limit of (200° F) (ref. 1, 2):

- NTI-100, NTI-101, Selected Incore Temperature or Temporary Thermocouples
- NTR-210, Reactor Coolant T-Cold Wide Range Loop 1
- NTR-220, Reactor Coolant T-Cold Wide Range Loop 2
- NTR-230, Reactor Coolant T-Cold Wide Range Loop 3
- NTR-240, Reactor Coolant T-Cold Wide Range Loop 4
- NTR-110, Reactor Coolant T-Hot Loop 1
- NTR-120, Reactor Coolant T-Hot Loop 2
- NTR-130, Reactor Coolant T-Hot Loop 3
- NTR-140, Reactor Coolant T-Hot Loop 4
- RHR display on PPC

In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on time to boil data (ref.2).

This IC addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit~~, or the inability to determine RCS temperature and level,~~ and represents a potential degradation of the level of safety of the plant. If the ~~RCS~~ RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the ~~Emergency Director~~ SEC should also refer to IC CA3.

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

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

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

~~EAL #2 reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.~~

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

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

CNP Basis Reference(s):

1. 1(2)-OHP-4021-001-004, Plant Cooldown from Hot Standby to Cold Shutdown
2. 1(2)-OHP-4022-017-001, Loss of RHR Cooling
3. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

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

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

Mode Applicability:

5 - Cold Shutdown, 6- Refueling

Definition(s):

None

Basis:

The following instrumentation is capable of providing indication of an RCS temperature rise that approaches the Technical Specification cold shutdown temperature limit of (200° F) (ref. 1, 2):

- NTI-100, NTI-101, Selected Incore Temperature or Temporary Thermocouples
- NTR-210, Reactor Coolant T-Cold Wide Range Loop 1
- NTR-220, Reactor Coolant T-Cold Wide Range Loop 2
- NTR-230, Reactor Coolant T-Cold Wide Range Loop 3
- NTR-240, Reactor Coolant T-Cold Wide Range Loop 4
- NTR-110, Reactor Coolant T-Hot Loop 1
- NTR-120, Reactor Coolant T-Hot Loop 2
- NTR-130, Reactor Coolant T-Hot Loop 3
- NTR-140, Reactor Coolant T-Hot Loop 4
- RHR display on PPC

RCS level indications include pressurizer level, narrow and wide range RVLIS and RC Loop narrow, mid and wide range instruments, NGG-100 and Mansell level instrument (ref. 2,3,4).

This ~~IC~~ EAL addresses ~~an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit, or~~ the inability to determine RCS temperature and level, and represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the ~~Emergency Director~~SEC should also refer to IC CA3.

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

~~EAL #1 involves a loss of decay heat removal capability, or an addition of heat to the RCS in excess of that which can currently be removed, such that reactor coolant temperature~~

~~cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.~~

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

~~EAL #2~~This EAL reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

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

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

CNP Basis Reference(s):

1. 1(2)-OHP-4021-001-004, Plant Cooldown from Hot Standby to Cold Shutdown
2. 1(2)-OHP-4022-017-001, Loss of RHR Cooling
3. 1(2)-OHP-4022-002-020 Excessive Reactor Coolant Leakage
4. 1(2)-OHP-4021-002-005, RCS Draining
5. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.1 Alert

UNPLANNED increase in RCS temperature to > 200°F for > Table C-4 duration (Notes 1, 10)

OR

UNPLANNED RCS pressure increase > 10 psig (This EAL does not apply during water-solid plant conditions

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

Note 10: Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss of decay heat removal.

Table C-4: RCS Heat-up Duration Thresholds

RCS Status	CONTAINMENT CLOSURE Status	Heat-up Duration
INTACT (but **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 – Refueling

Definition(s):

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

As applied to CNP, Containment Closure is established when the requirements of PMP-4100-SDR-001 are met.

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

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

REDUCED INVENTORY - Operating condition when fuel is in the reactor vessel and Reactor Coolant System level is lower than 3 feet (or more) below the Reactor Vessel flange.

Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1). These include (ref. 2, 3):

- NTI-100, NTI-101, Selected Incore Temperature or Temporary Thermocouples
- NTR-210, Reactor Coolant T-Cold Wide Range Loop 1
- NTR-220, Reactor Coolant T-Cold Wide Range Loop 2
- NTR-230, Reactor Coolant T-Cold Wide Range Loop 3
- NTR-240, Reactor Coolant T-Cold Wide Range Loop 4
- NTR-110, Reactor Coolant T-Hot Loop 1
- NTR-120, Reactor Coolant T-Hot Loop 2
- NTR-130, Reactor Coolant T-Hot Loop 3
- NTR-140, Reactor Coolant T-Hot Loop 4
- RHR display on PPC

The following instrumentation is capable of providing indication of a 10 psig rise in RCS pressure:

- NLI-1000A/B, RCS Pressure
- NLI-122A/B (MLMS Cart C), RCS Pressure
- NPS-110 (Loop 1) Reactor Vessel Train A Wide Range Pressure
- NPS-111 (Loop 3) Reactor Vessel Train B Wide Range Pressure

In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the RCS pressure increase criteria when in Mode 5 or based on time to boil data when in Mode 6 (ref. 3).

This IC addresses conditions involving a loss of decay heat removal capability or an addition of heat to the RCS in excess of that which can currently be removed. Either condition represents an actual or potential substantial degradation of the level of safety of the plant.

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

The ~~RCS~~ RCS Heat-up Duration Thresholds table addresses an increase in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not INTACT, or RCS inventory is reduced (e.g., mid-loop operation ~~in PWRs~~). The 20-minute criterion was included to allow time for operator action to address the temperature increase.

The RCS Heat-up Duration Thresholds table also addresses an increase in RCS temperature with the RCS INTACT. The status of CONTAINMENT CLOSURE is not crucial in this condition since the intact RCS is providing a high pressure barrier to a fission product release. The 60-minute time frame should allow sufficient time to address the temperature increase without a substantial degradation in plant safety.

Finally, in the case where there is an increase in RCS temperature, the RCS is not INTACT or is at REDUCED INVENTORY ~~[*PWR*]~~, and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because 1) the evaporated reactor coolant may be released directly into the containment atmosphere and subsequently to the environment, and 2) there is reduced reactor coolant inventory above the top of irradiated fuel.

~~EAL #2~~The RCS pressure increase threshold provides a pressure-based indication of RCS heat-up.

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

CNP Basis Reference(s):

1. CNP Technical Specifications Table 1.1-1
2. 1(2)-OHP-4021-001-004, Plant Cooldown from Hot Standby to Cold Shutdown
3. 1(2)-OHP-4022-017-001, Loss of RHR Cooling
4. NEI 99-01 CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – Loss of Vital DC Power

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

EAL:

CU4.1 Unusual Event

< 215 VDC bus voltage indications on Technical Specification **required** 250 VDC vital buses for ≥ 15 min. (Note 1)

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

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

None

Basis:

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. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss. The fifteen minute interval is intended to exclude transient or momentary power losses.

The vital DC buses are the following 250 VDC Class 1E buses (ref. 2, 3):

Train A:	Train B:
1CD (2CD)	1AB (2AB)

There are two, 116 cell, lead-acid storage batteries (1AB (2AB) and 1CD (2CD)) that supplement the output of the battery chargers. They supply DC power to the distribution buses when AC power to the chargers is lost or when transient loads exceed the capacity of the battery chargers (ref. 3).

CNP Technical Specification LCO 3.8.5 requires that one Train A or Train B 250 VDC electrical power subsystem shall be OPERABLE to support one train of the DC Electrical Power Distribution System required by LCO 3.8.10, "Distribution Systems - Shutdown." (ref. 1).

Per SD-DCC-NEEP-104, a 210 VDC lower limit has been identified from the battery service test acceptance criteria. Based on interpolation, the low voltage limit that would provide a 15 minute margin has been determined to be 213 VDC (ref. 4).

An EAL value of 215 VDC has been selected to account for available instrument accuracy. Meter scaling on installed control room instrumentation (10 VDC divisions on a dial indicator) limits the closest value that can be accurately read on the control board to 5 VDC.

This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS7.1.

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

As used in this EAL, "required" means the vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if Train A is out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an UNUSUAL EVENT. A loss of Vital DC power to Train A would not warrant an emergency classification.

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

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

CNP Basis Reference(s):

1. Technical Specifications Section 3.8.5 DC Sources - Shutdown
2. UFSAR Figure 8.3-2
3. UFSAR Section 8.3.4 250 Volt DC System (Safety Related)
4. SD-DCC-NEEP-104 250 VDC System
5. NEI 99-01 CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.1 Unusual Event

Loss of **all** Table C-5 onsite communication methods

OR

Loss of **all** Table C-5 ORO communication methods

OR

Loss of **all** Table C-5 NRC communication methods

Table C-5 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Plant Page	X		
Plant Radios	X	X	
Plant Telephone	X	X	X
ENS Line		X	X
Commercial Telephone		X	X
Microwave Transmission		X	X

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, D – Defueled

Definition(s):

None

Basis:

Onsite/offsite communications include one or more of the systems listed in Table C-5 (ref. 1).

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

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

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

~~EAL #1~~The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

~~EAL #2~~The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are ~~(see Developer Notes))~~ the State and Berrien County EOCs~~.~~

~~EAL #3~~The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

CNP Basis Reference(s):

1. CNP Plant Emergency Plan Section F Emergency Communications
2. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting Safety Systems

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

EAL:

CA6.1 Alert

The occurrence of **any** Table C-6 hazardous event

AND EITHER:

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

Table C-6 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the SEC

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

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

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

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

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

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

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

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

VISIBLE DAMAGE - Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did **not** cause damage to a structure or any other equipment) is **not** visible damage.

Basis:

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

~~EAL 1.b.1~~The first conditional addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

~~EAL 1.b.2~~The second conditional addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

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

CNP Basis Reference(s):

1. NEI 99-01 CA6

Category H – Hazards and Other Conditions Affecting Plant Safety

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

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

1. Security

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

2. Seismic Event

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

3. Natural or Technology Hazard

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

4. Fire

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

5. Hazardous Gas

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

6. Control Room Evacuation

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

7. SEC 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 SEC the latitude to classify emergency conditions consistent with the established classification criteria based upon SEC judgment.

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1 Unusual Event

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

OR

Notification of a credible security threat directed at the site

OR

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

Mode Applicability:

All

Definition(s):

SECURITY CONDITION - Any security event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A security condition does not involve a hostile action.

HOSTILE ACTION - An act toward CNP 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 CNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

Basis:

The security shift supervision is defined as the Security Shift Supervisor.

This EAL is based on the Donald C. Cook Nuclear Plant Security Plan (ref. 1).

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, and HS1 ~~and HG1~~.

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

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

~~EAL #1~~The first threshold references ~~(site-specific~~ the ~~securityshift supervision~~Shift Security Supervisor because these are the individuals trained to confirm that a security event is occurring or has occurred (ref. 1). Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.39 information.

~~EAL #2~~The second threshold addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with ~~(site-specific procedure)~~the CNP Plant Security Plan and DBT.

~~EAL #3~~The third threshold addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with Donald C. Cook Nuclear Plant Security Plan~~(site-specific procedure)~~.

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

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

CNP Basis Reference(s):

1. Donald C. Cook Nuclear Plant Security Plan (Safeguards)
2. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

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

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward CNP 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 CNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

OWNER CONTROLLED AREA - Area outside the PROTECTED AREA fence that immediately surrounds the plant. Access to this area is generally restricted to those entering on official business.

Basis:

The security shift supervision is defined as the Security Shift Supervisor (ref. 1).

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

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

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

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The ALERT declaration will also heighten the awareness of Offsite Response

Organizations (OROs), allowing them to be better prepared should it be necessary to consider further actions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

~~EAL #1~~The first threshold is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes any action directed against an ISFSI that is located outside the plant PROTECTED AREA.

~~EAL #2~~The second threshold addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and OROs are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with ~~(~~site-specific security procedures~~)~~.

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

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

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

CNP Basis Reference(s):

1. Donald C. Cook Nuclear Plant Security Plan (Safeguards)
2. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the plant PROTECTED AREA

EAL:

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

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward CNP 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 CNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

PROTECTED AREA - The area encompassed by physical barriers to control access to the plant and to the ISFSI.

Basis:

The security shift supervision is defined as the Security Shift Supervisor (ref. 1).

These individuals are the designated onsite personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Donald C. Cook Nuclear Plant Security Plan (Safeguards) information.

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

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3, 4 5).

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

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The SITE AREA EMERGENCY declaration will mobilize Offsite Response Organization (ORO) resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

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

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

Escalation of the emergency classification level would be via IC ~~HG1~~FG1.

CNP Basis Reference(s):

1. Donald C. Cook Nuclear Plant Security Plan (Safeguards)
2. NEI 99-01 HS1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE level

EAL:

HU2.1	**Unusual Event**
Control Room personnel feel an actual or potential seismic event **AND** The occurrence of a seismic event is confirmed in manner deemed appropriate by the Shift Manager	

Mode Applicability:

All

Definition(s):

None

Basis:

Ground motion acceleration of 0.10g horizontal is the Operating Basis Earthquake for CNP (ref. 1).

To avoid inappropriate emergency classification resulting from spurious actuation of the seismic instrumentation or felt motion not attributable to seismic activity, an offsite agency (USGS, National Earthquake Information Center) can confirm that an earthquake has occurred in the area of the plant. 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 CNP. Alternatively, near real-time seismic activity can be accessed via the NEIC website:

http://earthquake.usgs.gov

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE). An earthquake greater than an OBE but less than a Safe Shutdown Earthquake (SSE) should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., performs walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event (e.g., lateral accelerations in excess of 0.08g). The Shift Manager or Site Emergency ~~Director~~ Coordinator may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration (ref. 2).

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

1(2)-OHP-4022-001-007 Earthquake provides the guidance for determining if the OBE earthquake threshold is exceeded and any required response actions. (ref. 2). Because CNP seismic instrumentation does not provide direct and timely indications of having exceeded the OBE ground acceleration, the alternative EAL wording specified in the generic NEI 99-01 HU2 developers note (felt earthquake) is implemented.

CNP Basis Reference(s):

1. FSAR Section 1.3.1 Structures and Equipment
2. 1(2)-OHP-4022-001-007 Earthquake
3. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

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

Mode Applicability:

All

Definition(s):

PROTECTED AREA - The area encompassed by physical barriers to control access to the plant and to the ISFSI.

Basis:

Response actions associated with a tornado onsite is provided in 12-OHP-4022-001-010 Severe Weather (ref. 1).

If damage is confirmed visually or by other in-plant indications, the event may be escalated to an ALERT under EAL CA6.1 or SA9.1.

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

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

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

~~EAL #2 addresses flooding of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.~~

~~EAL #3 addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.~~

~~EAL #4 addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site flooding caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.~~

~~This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.~~

~~EAL #5 addresses (site-specific description).~~

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

CNP Basis Reference(s):

1. 12-OHP-4022-001-010 Severe Weather
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.2	**Unusual Event**
Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode	

Mode Applicability:

All

Definition(s):

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

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

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

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

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

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

Basis:

Refer to EALs CA6.1 or SA9.1 for internal flooding affecting one or more SAFETY SYSTEM trains.

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

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

This EAL addresses FLOODING of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

~~EAL #3 addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.~~

~~EAL #4 addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site flooding caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.~~

~~This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.~~

~~EAL #5 addresses (site-specific description).~~

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

CNP Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.3	**Unusual Event**
Movement of personnel within the plant PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)	

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - The area encompassed by physical barriers to control access to the plant and to the ISFSI.

Basis:

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

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

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

~~This EAL addresses flooding of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.~~

~~EAL #3~~This EAL addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.

~~EAL #4 addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site flooding caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.~~

~~This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane~~

~~Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.~~

~~EAL #5 addresses (site-specific description).~~

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

CNP Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.~~EAL #1 addresses a tornado striking (touching down) within the PROTECTED AREA.~~

~~This EAL addresses flooding of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.~~

~~EAL #3 addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.~~

~~EAL #4~~This EAL addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site FLOODING caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.

This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.

~~EAL #5 addresses (site-specific description).~~ Escalation of the emergency classification level would be based on ICs in Recognition Categories ~~A~~R, F, S or C.

CNP Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.1 Unusual Event

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

- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

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

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

Table H-1 Fire Areas
• Control Room
• Containment
• Auxiliary Building
• Switchgear Areas
• Diesel Generator Rooms
• ESW System enclosures
• AFW Pump Rooms
• Refueling Water Storage Tank
• Condensate Storage Tank

Mode Applicability:

All

Definition(s):

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

Basis:

Table H-1 Fire Areas are based on Fire Hazards Analysis Units No. 1 and 2. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1).

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

~~EAL #1~~

~~The~~ For EAL HU4.1 the intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.~~EAL #2~~

~~This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.~~

~~A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.~~

~~If an actual FIRE is verified by a report from the field, then EAL #1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.~~

~~EAL #3~~

~~In addition to a FIRE addressed by EAL #1 or EAL #2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety. This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA. [Sentence for plants with an ISFSI outside the plant Protected Area]~~

~~EAL #4~~

~~If a FIRE within the plant or ISFSI [for plants with an ISFSI outside the plant Protected Area] PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the~~

~~Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.~~

~~Basis-Related Requirements from Appendix R~~

~~Appendix R to 10 CFR 50, states in part:~~

> ~~Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."~~
>
> ~~When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.~~
>
> ~~Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.~~

~~In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.~~

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

CNP Basis Reference(s):

1. Fire Hazards Analysis Units No. 1 and 2
2. NEI 99-01 HU4

ATTACHMENT 1
EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.2	**Unusual Event**
Receipt of a single fire alarm (i.e., **no** other indications of a FIRE) **AND** The fire alarm is indicating a FIRE within **any** Table H-1 area **AND** The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1)	

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

Table H-1 Fire Areas
• Control Room • Containment • Auxiliary Building • Switchgear Areas • Diesel Generator Rooms • ESW System enclosures • AFW Pump Rooms • Refueling Water Storage Tank • Condensate Storage Tank

Mode Applicability:

All

Definition(s):

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

Basis:

<u>Table H-1 Fire Areas are based on Fire Hazards Analysis Units No. 1 and 2. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1).</u>

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

~~EAL #1~~

~~The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.~~

~~Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.~~

~~EAL #2~~

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then ~~EAL #1~~HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

~~EAL #3~~

~~In addition to a FIRE addressed by EAL #1 or EAL #2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety.~~ ~~*This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA.*~~ ~~[Sentence for plants with an ISFSI outside the plant Protected Area~~]

~~EAL #4~~

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

Basis-Related Requirements from Appendix R

(Note: CNP is not an Appendix R plant. This bases is cited only to justify the 30 minute timing component related to a single fire alarm)

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."
>
> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.
>
> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

CNP Basis Reference(s):

1. Fire Hazards Analysis Units No. 1 and 2
2. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

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

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

Mode Applicability:

All

Definition(s):

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

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.

PROTECTED AREA - The area encompassed by physical barriers to control access to the plant and to the ISFSI.

Basis:

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

~~EAL #1~~

~~The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.~~

~~Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.~~

~~EAL #2~~

~~This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the~~

~~30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.~~

~~A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.~~

~~If an actual FIRE is verified by a report from the field, then EAL #1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.~~

~~EAL #3~~

In addition to a FIRE addressed by EAL #~~1~~HU4.1 or ~~EAL #2~~HU4.2, a FIRE within the ~~plant~~ PROTECTED AREA (plant or ISFSI) not extinguished within 60-minutes may also potentially degrade the level of plant safety. This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA.~~[*Sentence for plants with an ISFSI outside the plant Protected Area*]~~

~~EAL #4~~

~~Basis-Related Requirements from Appendix R~~

~~Appendix R to 10 CFR 50, states in part:~~

> ~~Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."~~
>
> ~~When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.~~
>
> ~~Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.~~

~~In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.~~

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

CNP Basis Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

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

Mode Applicability:

All

Definition(s):

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

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.

PROTECTED AREA - The area encompassed by physical barriers to control access to the plant and to the ISFSI.

Basis:

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

~~EAL #1~~

~~The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.~~

~~Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.~~

~~EAL #2~~

~~This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.~~

~~A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.~~

~~If an actual FIRE is verified by a report from the field, then EAL #1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.~~

~~EAL #3~~

~~In addition to a FIRE addressed by EAL #1 or EAL #2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety.~~ ~~*This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA.*~~ ~~[*Sentence for plants with an ISFSI outside the plant Protected Area*]~~

~~EAL #4~~

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

~~Basis-Related Requirements from Appendix R~~

~~Appendix R to 10 CFR 50, states in part:~~

> ~~Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."~~
>
> ~~When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.~~
>
> ~~Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.~~

~~In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one~~

~~redundant train (G.2.c). As used in EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.~~

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

CNP Basis Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gases

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

EAL:

HA5.1 Alert

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

AND

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

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

Table H-2 Safe Operation & Shutdown Room/Areas

Room/Area	Mode Applicability
Auxiliary Building 573'	4, 5
Auxiliary Building 587' (including D/G room)	1, 2, 3, 4, 5
Auxiliary Building 591'	1, 2, 3, 4
Auxiliary Building 609' (including 4kV room)	1, 2, 3, 4, 5
Auxiliary Building 633'	1, 2, 3, 4
Turbine Building (**All** Levels)	1, 2, 3
Turbine Building 591'	4, 5
Screenhouse	1, 2, 3, 4, 5

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

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

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

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

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

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

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

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

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

CNP Basis Reference(s):

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

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

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

EAL:

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

Mode Applicability:

All

Definition(s):

None

Basis:

Plant control is considered to have been transferred when either 1) control of the plant is no longer maintained in the Control Room or 2) the last licensed operator has left the Control Room, whichever comes first.

The Shift Manager (SM) determines if the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions. 1(2)-OHP-4025-001-001, Emergency Remote Shutdown (ERS), and 1(2)-OHP-4022-CRE-001 Control Room Evacuation provides the instructions for tripping the unit, and maintaining RCS inventory and Hot Shutdown conditions from outside the Control Room (Ref. 1, 2).

This EAL is only applicable when the decision has been made to evacuate the Control Room, not when conditions are being evaluated.

Inability to establish plant control from outside the Control Room escalates this event to a SITE AREA EMERGENCY per EAL HS6.1.

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Escalation of the emergency classification level would be via IC HS6.

CNP Basis Reference(s):

1. 1(2)-OHP-4025-001-001, Emergency Remote Shutdown (ERS)
2. 1(2)-OHP-4022-CRE-001 Control Room Evacuation
3. NEI 99-01 HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

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

EAL:

HS6.1 Site Area Emergency

An event has resulted in plant control being transferred from the Control Room to the Local Shutdown Instrumentation

AND

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

- Reactivity Control (modes 1, 2 and 3 **only**)
- Core Cooling
- RCS heat removal

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown, 5 – Cold Shutdown, 6 - Refueling

Definition(s):

None

Basis:

The 15-minute time period starts when either 1) control of the plant is no longer maintained in the Control Room or 2) the last licensed operator has left the Control Room, whichever comes first.

The Shift Manager determines if the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions. 1(2)-OHP-4025-001-001, Emergency Remote Shutdown (ERS) and 1(2)-OHP-4022-CRE-001 Control Room Evacuation provides the instructions for tripping the unit, and maintaining RCS inventory and Hot Shutdown conditions from outside the Control Room (ref. 1, 2).

The intent of this EAL is to capture events in which control of the plant cannot be reestablished in a timely manner. The fifteen minute time for transfer starts when the last licensed operator has left the Control Room (not when 1(2)-OHP-4025-001-001 or 1(2)-OHP-4022-CRE-001 is entered). The time interval is based on how quickly control must be reestablished without core uncovery and/or core damage. The determination of whether or not control is established from outside the Control Room is based on SEC judgment. The SEC is expected to make a reasonable, informed judgment that control of the plant from outside the Control Room cannot be established within the fifteen minute interval.

Once the Control Room is evacuated, the objective is to establish control of important plant equipment and maintain knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on 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), RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink).

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

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on Site Emergency ~~Director~~ Coordinator judgment. The Site Emergency ~~Director~~ Coordinator is expected to make a reasonable, informed judgment within ~~(the site-specific time for transfer)~~15 minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

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

CNP Basis Reference(s):

1. 1(2)-OHP-4025-001-001, Emergency Remote Shutdown (ERS)
2. 1(2)-OHP-4022-CRE-001 Control Room Evacuation
3. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

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

EAL:

HU7.1	**Unusual Event**
Other conditions exist which in the judgment of the SEC indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.	

Mode Applicability:

All

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

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

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

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

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

Basis:

The SEC is the designated onsite individual having the responsibility and authority for implementing the CNP Emergency Plan (ref. 1). The Shift Manager (SM) initially acts in the capacity of the SEC and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the SEC, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

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

CNP Basis Reference(s):

1. CNP Emergency Plan section B.5.a.1 Site Emergency Coordinator
2. CNP Emergency Plan section B.1.k On-Shift Operations Personnel
3. NEI 99-01 HU7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

Initiating Condition: Other conditions exist that in the judgment of the SEC warrant declaration of an ALERT

EAL:

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

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward CNP 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 CNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

Basis:

The SEC is the designated onsite individual having the responsibility and authority for implementing the CNP Emergency Plan (ref. 1). The Shift Manager (SM) initially acts in the capacity of the SEC and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the SEC, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

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

CNP Basis Reference(s):

1. CNP Emergency Plan section B.5.a.1 Site Emergency Coordinator
2. CNP Emergency Plan section B.1.k On-Shift Operations Personnel
3. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

Initiating Condition: Other conditions existing that in the judgment of the SEC warrant declaration of a SITE AREA EMERGENCY

EAL:

HS7.1	**Site Area Emergency**
Other conditions exist which in the judgment of the SEC indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward CNP 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 CNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area)

Basis:

The SEC is the designated onsite individual having the responsibility and authority for implementing the CNP Emergency Plan (ref. 1). The Shift Manager (SM) initially acts in the capacity of the SEC and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the SEC, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

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

CNP Basis Reference(s):

1. CNP Emergency Plan section B.5.a.1 Site Emergency Coordinator
2. CNP Emergency Plan section B.1.k On-Shift Operations Personnel
3. NEI 99-01 HS7

ATTACHMENT 1
EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

Initiating Condition: Other conditions exist which in the judgment of the SEC warrant declaration of a GENERAL EMERGENCY

EAL:

HG7.1	**General Emergency**
Other conditions exist which in the judgment of the SEC indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward CNP 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 CNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

IMMINENT - The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

Basis:

The SEC is the designated onsite individual having the responsibility and authority for implementing the CNP Emergency Plan (ref. 1). The Shift Manager (SM) initially acts in the capacity of the SEC and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the SEC, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

Releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the site boundary.

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Site

Emergency ~~Director~~ Coordinator to fall under the emergency classification level description for a GENERAL EMERGENCY.

CNP Basis Reference(s):

1. CNP Emergency Plan section B.5.a.1 Site Emergency Coordinator
2. CNP Emergency Plan section B.1.k On-Shift Operations Personnel
3. NEI 99-01 HG7

Category S – System Malfunction

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

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

The events of this category pertain to the following subcategories:

1. Loss of Emergency AC Power

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

2. Loss of Vital DC Power

Loss of emergency electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of vital plant 250 VDC power sources.

3. Loss of Control Room Indications

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

4. RCS Activity

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

5. RCS Leakage

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

6. RPS Failure

This subcategory includes events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RPS to complete a reactor trip comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean any 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.

7. Loss of Communications

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

8. Containment Failure

Failure of containment isolation capability (under conditions in which the containment is not currently challenged) warrants emergency classification. Failure of containment pressure control capability also warrants emergency classification.

9. Hazardous Event Affecting Safety Systems

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

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SU1.1 **Unusual Event**
Loss of **all** offsite AC power capability, Table S-1, to emergency 4.16KV buses T11A (T21A) and T11D (T21D) for ≥ 15 min. (Note 1)

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

Table S-1 AC Power Sources
Offsite: • Unit Auxiliary Xmr TR1AB (TR2AB) • Unit Auxiliary Xmr TR1CD (TR2CD) • Reserve Auxiliary Xmr TR101AB (TR201AB) • Reserve Auxiliary Xmr TR101CD (TR201CD) • 69/4.16 kV Alternate Xmr TR12EP-1 **Onsite:** • EDG 1AB (2AB) • EDG 1CD (2CD)

Mode Applicability:

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

Definition(s):

None

Basis:

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

A list of offsite AC power sources credited for this EAL are specified in Table S-1.

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. 4.16KV buses T11A (T21A) and T11D (T21D) are the emergency (essential) buses (ref. 1). While generating, auxiliary power is normally supplied from the generator terminals through the unit auxiliary transformers (TR1AB and TR1CD for Unit 1 and TR2AB and TR2CD for Unit 2). When the plant trips or the plant is shutdown the station auxiliaries are transferred to the preferred offsite power source (that is, to reserve auxiliary transformers TR101AB and

TR101CD for Unit 1 and TR201AB and TR201CD for Unit 2) to assure continued power to equipment when the main generator is off-line (ref. 1, 2, 3).

In addition, an alternate offsite power source, a 69/4.16kV transformer (TR12EP-1), located at the plant site, has the necessary capacity to operate one train of the engineered safeguard equipment in one unit while supplying one train of the safe shutdown power in the other.

T11A (T21A) and T11D (T21D) also each have an emergency diesel generator which supply onsite electrical power to the bus automatically in the event that the preferred offsite sources become unavailable (ref. 1, 2, 3).

The Supplemental Diesel Generators (SDGs) are not credited as an AC power source for this EAL.

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

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

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

Escalation of the emergency classification level would be via IC SA1.

CNP Basis Reference(s):

1. UFSAR Figure 8.1-1A(B) Main Auxiliary One-Line Diagram
2. UFSAR Section 8.0 Electrical Systems
3. 1(2)-OHP-4022-001-005 Loss of Offsite Power with Reactor Shutdown
4. NEI 99-01 SU1

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SA1.1 **Alert**
AC power capability, Table S-1, to emergency 4.16KV buses T11A (T21A) and T11D (T21D) reduced to a single power source for ≥ 15 min. (Note 1)
AND
Any additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS

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

Table S-1 AC Power Sources
Offsite: • Unit Auxiliary Xmr TR1AB (TR2AB) • Unit Auxiliary Xmr TR1CD (TR2CD) • Reserve Auxiliary Xmr TR101AB (TR201AB) • Reserve Auxiliary Xmr TR101CD (TR201CD) • 69/4.16 kV Alternate Xmr TR12EP-1 **Onsite:** • EDG 1AB (2AB) • EDG 1CD (2CD)

Mode Applicability:

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

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

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

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

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

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

Basis:

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

A list of onsite and offsite AC power sources credited for this EAL are specified in Table S-1.

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. 4.16KV buses T11A (T21A) and T11D (T21D) are the emergency (essential) buses (ref. 1). While generating, auxiliary power is normally supplied from the generator terminals through the unit auxiliary transformers (TR1AB and TR1CD for Unit 1 and TR2AB and TR2CD for Unit 2). When the plant trips or the plant is shutdown the station auxiliaries are transferred to the preferred offsite power source (that is, to reserve auxiliary transformers TR101AB and TR101CD for Unit 1 and TR201AB and TR201CD for Unit 2) to assure continued power to equipment when the main generator is off-line (ref. 1, 2, 3).

In addition, an alternate offsite power source, a 69/4.16kV transformer (TR12EP-1), located at the plant site, has the necessary capacity to operate one train of the engineered safeguard equipment in one unit while supplying one train of the safe shutdown power in the other.

T11A (T21A) and T11D (T21D) also each have an emergency diesel generator which supply onsite electrical power to the bus automatically in the event that the preferred offsite sources become unavailable (ref. 1, 2, 3).

The Supplemental Diesel Generators (SDGs) are not credited as an AC power source for this EAL.

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

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

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

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

Escalation of the emergency classification level would be via IC SS1.

CNP Basis Reference(s):

1. UFSAR Figure 8.1-1A(B) Main Auxiliary One-Line Diagram
2. UFSAR Section 8.0 Electrical Systems

3. 1(2)-OHP-4022-001-005 Loss of Offsite Power with Reactor Shutdown
4. NEI 99-01 SA1

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SS1.1	**Site Area Emergency**
Loss of **all** offsite and **all** onsite AC power to emergency 4.16KV buses T11A (T21A) and T11D (T21D) for ≥ 15 min. (Note 1)	

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

Mode Applicability:

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

Definition(s):

None

Basis:

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. 4.16KV buses T11A (T21A) and T11D (T21D) are the emergency (essential) buses (ref. 1). While generating, auxiliary power is normally supplied from the generator terminals through the unit auxiliary transformers (TR1AB and TR1CD for Unit 1 and TR2AB and TR2CD for Unit 2). When the plant trips or the plant is shutdown the station auxiliaries are transferred to the preferred offsite power source (that is, to reserve auxiliary transformers TR101AB and TR101CD for Unit 1 and TR201AB and TR201CD for Unit 2) to assure continued power to equipment when the main generator is off-line (ref. 1, 2, 3).

In addition, an alternate offsite power source, a 69/4.16kV transformer (TR12EP-1), located at the plant site, has the necessary capacity to operate one train of the engineered safeguard equipment in one unit while supplying one train of the safe shutdown power in the other.

T11A (T21A) and T11D (T21D) also each have an emergency diesel generator which supply onsite electrical power to the bus automatically in the event that the preferred offsite sources become unavailable (ref. 1, 2, 3).

The Supplemental Diesel Generators (SDGs) or any other alternative AC power source capable of powering an emergency bus can also be credited as an AC power source for this EAL.

The 15-minute interval begins when both offsite and onsite AC power capability are lost (ref. 4).

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs ~~AG1~~RG1, FG1 or SG1.

CNP Basis Reference(s):

1. UFSAR Figure 8.1-1A(B) Main Auxiliary One-Line Diagram
2. UFSAR Section 8.0 Electrical Systems
3. 1(2)-OHP-4022-001-005 Loss of Offsite Power with Reactor Shutdown
4. 1(2)-OHP-4023-ECA-0.0 Loss of All AC Power
5. NEI 99-01 SS1

ATTACHMENT 1
EAL Bases

Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SG1.1	**General Emergency**

Loss of **all** offsite and **all** onsite AC power to emergency 4.16KV buses T11A (T21A) and T11D (T21D)

AND EITHER:

- Restoration of at least one emergency bus in < 4 hours is **not** likely (Note 1)
- CSFST Core Cooling RED Path (F.0-2) conditions met

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

Mode Applicability:

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

Definition(s):

None

Basis:

This EAL is indicated by the extended loss of all offsite and onsite AC power capability to 4.16KV emergency buses T11A (T21A) and T11D (T21D) either for greater then the CNP Station Blackout (SBO) coping analysis time (4 hrs.) (ref. 1, 4) or that has resulted in indications of an actual loss of adequate core cooling.

Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED PATH conditions being met. (ref. 5).

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. 4.16KV buses T11A (T21A) and T11D (T21D) are the emergency (essential) buses (ref. 1). While generating, auxiliary power is normally supplied from the generator terminals through the unit auxiliary transformers (TR1AB and TR1CD for Unit 1 and TR2AB and TR2CD for Unit 2). When the plant trips or the plant is shutdown the station auxiliaries are transferred to the preferred offsite power source (that is, to reserve auxiliary transformers TR101AB and TR101CD for Unit 1 and TR201AB and TR201CD for Unit 2) to assure continued power to equipment when the main generator is off-line (ref. 1, 2, 3).

In addition, an alternate offsite power source, a 69/4.16kV transformer (TR12EP-1), located at the plant site, has the necessary capacity to operate one train of the engineered safeguard equipment in one unit while supplying one train of the safe shutdown power in the other.

T11A (T21A) and T11D (T21D) also each have an emergency diesel generator which supply onsite electrical power to the bus automatically in the event that the preferred offsite sources become unavailable (ref. 1, 2, 3).

The Supplemental Diesel Generators (SDGs) or any other alternative AC power source capable of powering an emergency bus can also be credited as an AC power source for this EAL.

Four hours is the station blackout coping time (ref. 4).

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on SEC judgment as it relates to imminent Loss of fission product barriers and degraded ability to monitor fission product barriers. Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED PATH conditions being met. Specifically, Core Cooling RED PATH conditions exist if either the five highest core exit TCs are reading greater than or equal to 1200ºF or core exit TCs are reading greater than or equal to 757ºF with RCS subcooling less than or equal to 40ºF, and RVLIS indication is less than or equal to that specified based on the number of RCPs running (ref. 5).

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

The EAL should require declaration of a GENERAL EMERGENCY prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from SITE AREA EMERGENCY will occur if it is projected that power cannot be restored to at least one AC emergency bus by the end of the analyzed station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of challenges to multiple fission product barriers.

The estimate for restoring at least one emergency bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, protective actions for the public.

The EAL will also require a GENERAL EMERGENCY declaration if the loss of AC power results in parameters that indicate an inability to adequately remove decay heat from the core.

CNP Basis Reference(s):

1. UFSAR Figure 8.1-1A(B) Main Auxiliary One-Line Diagram
2. UFSAR Section 8.0 Electrical Systems
3. 1(2)-OHP-4023-ECA-0.0 Loss of All AC Power
4. UFSAR Section 8.7 Station Blackout
5. 1(2)-OHP-4023-F-0.2 Core Cooling

6. NEI 99-01 SG1

Category: S – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

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

EAL:

SS2.1	**Site Area Emergency**
Loss of **all** 250 VDC power based on bus voltage indications < 215 VDC on **all** vital DC buses 1CD (2CD) (Train A) and 1AB (2AB) (Train B) for ≥ 15 min. (Note 1)	

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

Mode Applicability:

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

Definition(s):

None

Basis:

The vital DC buses are the following 250 VDC Class 1E buses (ref. 1, 2, 3):

Train A:	Train B:
1CD (2CD)	1AB (2AB)

There are two, 116 cell, lead-acid storage batteries (1AB (2AB) and 1CD (2CD)) that supplement the output of the battery chargers. They supply DC power to the distribution buses when AC power to the chargers is lost or when transient loads exceed the capacity of the battery chargers (ref. 3).

CNP Technical Specification LCO 3.8.4 requires that both Train A and Train B 250 VDC electrical power subsystem shall be OPERABLE to support both trains of the DC Electrical Power Distribution System required by LCO 3.8.9, "Distribution Systems - Operating." (ref. 1).

Per SD-DCC-NEEP-104, a 210 VDC lower limit has been identified from the battery service test acceptance criteria. Based on interpolation, the low voltage limit that would provide a 15 minute margin has been determined to be 213 VDC (ref. 4).

An EAL value of 215 VDC has been selected to account for available instrument accuracy. Meter scaling on installed control room instrumentation (10 VDC divisions on a dial indicator) limits the closest value that can be accurately read on the control board to 5 VDC.

This IC addresses a loss of vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs ~~AG1~~RG1, FG1 or ~~SG8~~SG2.

CNP Basis Reference(s):

1. Technical Specifications Section 3.8.4 DC Sources - Operating
2. UFSAR Figure 8.3-2
3. UFSAR Section 8.3.4 250 Volt DC System (Safety Related)
4. SD-DCC-NEEP-104 250 VDC System
5. NEI 99-01 SS8

Category: S –System Malfunction

Subcategory: 2 – Loss of Vital DC Power

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

EAL:

SG2.1 General Emergency

Loss of **all** offsite and **all** onsite AC power to emergency 4.16KV buses T11A (T21A) and T11D (T21D) for ≥ 15 min.

AND

Loss of **all** 250 VDC power based on bus voltage indications < 215 VDC on **all** vital DC buses 1CD (2CD) (Train A) and 1AB (2AB) (Train B) for ≥ 15 min.

(Note 1)

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

Mode Applicability:

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

Definition(s):

None

Basis:

This EAL is indicated by the loss of all offsite and onsite emergency AC power capability to 4.16KV emergency buses T11A (T21A) and T11D (T21D) for greater than 15 minutes in combination with degraded vital DC power voltage. This EAL addresses operating experience from the March 2011 accident at Fukushima Daiichi.

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. 4.16KV buses T11A (T21A) and T11D (T21D) are the emergency (essential) buses (ref. 1). While generating, auxiliary power is normally supplied from the generator terminals through the unit auxiliary transformers (TR1AB and TR1CD for Unit 1 and TR2AB and TR2CD for Unit 2). When the plant trips or the plant is shutdown the station auxiliaries are transferred to the preferred offsite power source (that is, to reserve auxiliary transformers TR101AB and TR101CD for Unit 1 and TR201AB and TR201CD for Unit 2) to assure continued power to equipment when the main generator is off-line (ref. 1, 2, 3).

In addition, an alternate offsite power source, a 69/4.16kV transformer (TR12EP-1), located at the plant site, has the necessary capacity to operate one train of the engineered safeguard equipment in one unit while supplying one train of the safe shutdown power in the other.

T11A (T21A) and T11D (T21D) also each have an emergency diesel generator which supply onsite electrical power to the bus automatically in the event that the preferred offsite sources become unavailable (ref. 1, 2, 3).

The Supplemental Diesel Generators (SDGs) or any other alternative AC power source capable of powering an emergency bus can also be credited as an AC power source for this EAL.

The vital DC buses are the following 250 VDC Class 1E buses (ref. 4, 5, 6):

Train A:	Train B:
1CD (2CD)	1AB (2AB)

There are two, 116 cell, lead-acid storage batteries (1AB (2AB) and 1CD (2CD)) that supplement the output of the battery chargers. They supply DC power to the distribution buses when AC power to the chargers is lost or when transient loads exceed the capacity of the battery chargers (ref. 6).

CNP Technical Specification LCO 3.8.4 requires that both Train A and Train B 250 VDC electrical power subsystem shall be OPERABLE to support both trains of the DC Electrical Power Distribution System required by LCO 3.8.9, "Distribution Systems - Operating." (ref. 4).

Per SD-DCC-NEEP-104, a 210 VDC lower limit has been identified from the battery service test acceptance criteria. Based on interpolation, the low voltage limit that would provide a 15 minute margin has been determined to be 213 VDC (ref. 7).

An EAL value of 215 VDC has been selected to account for available instrument accuracy. Meter scaling on installed control room instrumentation (10 VDC divisions on a dial indicator) limits the closest value that can be accurately read on the control board to 5 VDC.

This IC addresses a concurrent and prolonged loss of both emergency AC and Vital DC power. A loss of all emergency AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both emergency AC and vital DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

CNP Basis Reference(s):

1. UFSAR Figure 8.1-1A(B) Main Auxiliary One-Line Diagram
2. UFSAR Section 8.0 Electrical Systems
3. 1(2)-OHP-4023-ECA-0.0 Loss of All AC Power

4. Technical Specifications Section 3.8.4 DC Sources - Operating
5. UFSAR Figure 8.3-2
6. UFSAR Section 8.3.4 250 Volt DC System (Safety Related)
7. SD-DCC-NEEP-104 250 VDC System
8. NEI 99-01 SG8

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer

EAL:

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

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

Table S-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core Exit TC temperature • Level in at least one SG • Auxiliary feed flow in at least one SG

Mode Applicability:

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

Definition(s):

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

Basis:

<u>SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Process Computer, which displays SPDS required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1).</u>

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an “inability to monitor” means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling ~~[PWR] / RPV level [BWR]~~ and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level ~~[PWR] / RPV water level [BWR]~~ cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via IC ~~SA2~~SA3.

CNP Basis Reference(s):

1. UFSAR Section 7.5 Engineered Safety Features Instrumentation
2. NEI 99-01 SU2

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress

EAL:

SA3.1	**Alert**

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

AND

Any significant transient is in progress, Table S-3

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

Table S-2 Safety System Parameters
• Reactor power
• RCS level
• RCS pressure
• Core Exit TC temperature
• Level in at least one SG
• Auxiliary feed flow in at least one SG

Table S-3 Significant Transients
• Reactor trip
• Runback ≥ 25% thermal power
• Electrical load rejection > 25% of full electrical load
• ECCS actuation

Mode Applicability:

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

Definition(s):

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

Basis:

SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Computer, which displays SPDS required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1).

Significant transients are listed in Table S-3 and include response to automatic or manually initiated functions such as reactor trips, runbacks involving greater than or equal to 25% thermal power change, electrical load rejections of greater than 25% full electrical load or ECCS (SI) injection actuations.

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling ~~[*PWR*] / RPV level [*BWR*]~~ and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level ~~[*PWR*] / RPV water level [*BWR*]~~ cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via ICs FS1 or IC ~~AS1~~RS1

CNP Basis Reference(s):

1. UFSAR Section 7.5 Engineered Safety Features Instrumentation
2. NEI 99-01 SA2

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.1	**Unusual Event**
Sample analysis indicates RCS activity > Technical Specification Section 3.4.16 limits	

Mode Applicability:

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

Definition(s):

None

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category ~~A~~ R ICs.

CNP Basis Reference(s):

1. CNP Technical Specifications section 3.4.16 RCS Specific Activity
2. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1	**Unusual Event**
RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. **OR** RCS identified leakage > 25 gpm for ≥ 15 min. **OR** Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min. (Note 1)	

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

Mode Applicability:

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

Definition(s):

None

Basis:

Manual or computer-based methods of performing an RCS inventory balance are normally used to determine RCS leakage (ref. 1).

Identified leakage includes

- Leakage such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank, or
- Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage, or
- RCS leakage through a steam generator to the secondary system (ref. 2).

Unidentified leakage is all leakage (except RCP seal water injection or leakoff) that is not identified leakage (ref. 2).

Pressure Boundary leakage is leakage (except SG leakage) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall (ref. 2)

RCS leakage outside of the containment that is not considered identified or unidentified leakage per Technical Specifications includes leakage via interfacing systems such as RCS to the Component Cooling Water, or systems that directly see RCS pressure outside containment such as Chemical & Volume Control System, Nuclear Sampling system and Residual Heat Removal system (when in the shutdown cooling mode) (ref. 3, 4)

Escalation of this EAL to the ALERT level is via Category F, Fission Product Barrier Degradation, EAL FA1.1.

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

~~EAL #1 and EAL #2~~The first and second EAL conditions are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications). ~~EAL #3~~The third condition addresses an RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These ~~EALs~~ conditions thus apply to leakage into the containment, a secondary-side system (e.g., steam generator tube leakage~~ in a PWR~~) or a location outside of containment.

The leak rate values for each ~~EAL~~ condition were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). ~~EAL #1~~ The first condition uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. ~~For PWRs, a~~An emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated). ~~For BWRs, a stuck-open Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.~~

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

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

CNP Basis Reference(s):

1. 1(2)-OHP-4030-102-016 Reactor Coolant System Leak Test
2. CNP Technical Specifications Definitions section 1.1
3. UFSAR Section 4.2.7 Leakage
4. 1(2)-OHP-4022-002-020 Excessive Reactor Coolant Leakage
5. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

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

EAL:

SU6.1 Unusual Event

An automatic trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** RPS setpoint is exceeded

AND

A subsequent automatic trip or manual trip action taken at the reactor control console (reactor trip switches) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

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

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) trip function. A reactor trip is automatically initiated by the RPS when certain continuously monitored parameters exceed predetermined setpoints (ref. 1, 2).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. 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 and a lowering of power into the source range. A successful trip has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2, 3, 4).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the reactor control console; manual reactor trip switches. Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as tripping the main turbine, locally opening reactor trip breakers, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 4).

Following any automatic RPS trip signal, E-0 (ref. 2) and /FR-S.1 (ref. 4) prescribe insertion of redundant manual trip signals to back up the automatic RPS trip function and ensure reactor shutdown is achieved. Even if the first subsequent manual trip signal inserts all control rods to the full-in position immediately after the initial failure of the automatic trip, the lowest level of classification that must be declared is an UNUSUAL EVENT (ref. 4).

A reactor trip resulting from actuation of the ATWS Mitigation System Actuation Circuitry (AMSAC) logic that results in full insertion of control rods and diminishing neutron flux is considered a successful reactor trip. AMSAC automatically initiates auxiliary feedwater and a turbine trip under conditions indicative of an Anticipated Transient Without Scram (ATWS) event (ref. 5).

In the event that the operator identifies a reactor trip is imminent and initiates a successful manual reactor trip before the automatic RPS trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. However, if subsequent manual reactor trip actions fail to reduce reactor power below 5%, the event escalates to the ALERT under EAL SA6.1.

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

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(~~trip ~~[*PWR*] / scram [*BWR*])~~that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic ~~(~~trip~~[*PWR*] / scram [*BWR*])~~ is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor ~~(~~trip, ~~[*PWR*] / scram [*BWR*~~operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor ~~(~~trip~~[*PWR*] / scram [*BWR*])~~). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor ~~(~~trip ~~[*PWR*] / scram [*BWR*])~~is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor ~~(~~trip~~[*PWR*] / scram [*BWR*])~~) using a different switch. Depending upon several factors, the initial or subsequent effort to manually ~~(~~trip~~[*PWR*] / scram [*BWR*])~~ the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor ~~(~~trip ~~[*PWR*] / scram [*BWR*])~~signal. If a subsequent manual or automatic ~~(~~trip ~~[*PWR*] / scram [*BWR*])~~is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor ~~(~~trip~~[*PWR*] / scram [*BWR*])~~). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

~~Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action. [BWR]~~

The plant response to the failure of an automatic or manual reactor ~~(~~trip ~~[PWR] / scram [BWR])~~will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an ALERT via IC ~~SA5~~SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC ~~SA5~~ SA6 or FA1, an UNUSUAL EVENT declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor ~~(~~trip ~~[PWR] / scram [BWR])~~signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor ~~(~~trip ~~[PWR] / scram [BWR])~~ and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the ~~(~~trip ~~[PWR] / scram [BWR])~~failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

CNP Basis Reference(s):

1. CNP Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. 1(2)-OHP04023-E-0 Reactor Trip or Safety Injection
3. 1(2)-OHP04023-F-0.1 Critical Safety Function Status Trees - Subcriticality
4. 1(2)-OHP-4023-FR-S-1 Response to Nuclear Power Generation/ATWS
5. UFSAR Section 3.3.3 Anticipated Transients Without Scram

6 NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

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

EAL:

SU6.2 Unusual Event

A manual trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** manual trip action was initiated

AND

A subsequent automatic trip or manual trip action taken at the reactor control console (reactor trip switches) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

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

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

This EAL addresses a failure of a manually initiated trip in the absence of having exceeded an automatic RPS trip setpoint and a subsequent automatic or manual trip is successful in shutting down the reactor (reactor power < 5%). (ref. 1).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. 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 and a lowering of power into the source range. A successful trip has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2, 3, 4).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the reactor control console; manual reactor trip switches. Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as tripping the main turbine, locally opening reactor trip breakers, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 4).

Following the failure of any manual trip signal, E-0 (ref. 2) and FR-S.1 (ref. 4) prescribe insertion of redundant manual trip signals to back up the RPS trip function and ensure reactor shutdown is achieved. Even if a subsequent automatic trip signal or the first subsequent manual trip signal inserts all control rods to the full-in position immediately after the initial failure of the manual trip, the lowest level of classification that must be declared is an UNUSUAL EVENT (ref. 4).

A reactor trip resulting from actuation of the ATWS Mitigation System Actuation Circuitry (AMSAC) logic that results in full insertion of control rods and diminishing neutron flux is considered a successful reactor trip. AMSAC automatically initiates auxiliary feedwater and a turbine trip under conditions indicative of an Anticipated Transient Without Scram (ATWS) event (ref. 5).

If both subsequent automatic and subsequent manual reactor trip actions in the Control Room fail to reduce reactor power below the power associated with the safety system design (< 5%) following a failure of an initial manual trip, the event escalates to an ALERT under EAL SA6.1.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(~~trip ~~[PWR] / scram [BWR])~~that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic ~~(~~trip~~[PWR] / scram [BWR])~~ is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor ~~(~~trip~~[PWR] / scram [BWR])~~, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor ~~(~~trip~~[PWR] / scram [BWR])~~). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor ~~(~~trip~~[PWR] / scram [BWR])~~ is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor ~~(~~trip~~[PWR] / scram [BWR])~~) using a different switch. Depending upon several factors, the initial or subsequent effort to manually trip~~(trip [PWR] / scram[BWR])~~ the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor ~~(~~trip ~~[PWR] / scram [BWR])~~signal. If a subsequent manual or automatic ~~(~~trip ~~[PWR] / scram [BWR])~~is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor ~~(~~trip~~[PWR] / scram [BWR])~~). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

~~Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action. [BWR]~~

The plant response to the failure of an automatic or manual reactor ~~(~~trip ~~[PWR] / scram [BWR])~~will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an ALERT via IC ~~SA5~~SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC ~~SA5~~ SA6 or FA1, an UNUSUAL EVENT declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor ~~(~~trip~~[PWR] / scram [BWR])~~ signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor ~~(~~trip ~~[PWR] / scram [BWR])~~ and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the ~~(~~trip ~~[PWR] / scram [BWR])~~failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

CNP Basis Reference(s):

1. CNP Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. 1(2)-OHP04023-E-0 Reactor Trip or Safety Injection
3. 1(2)-OHP04023-F-0.1 Critical Safety Function Status Trees - Subcriticality
4. 1(2)-OHP-4023-FR-S-1 Response to Nuclear Power Generation/ATWS
5. UFSAR Section 3.3.3 Anticipated Transients Without Scram
6. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control console are not successful in shutting down the reactor

EAL:

SA6.1 Alert

An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%

AND

Manual trip actions taken at the reactor control console (reactor trip switches) are **not** successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8)

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

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

This EAL addresses any automatic or manual reactor trip signal that fails to shut down the reactor (reactor power < 5%) followed by a subsequent manual trip that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed (ref. 1, 2).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the reactor control console; manual reactor trip switches. Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as tripping the main turbine, locally opening reactor trip breakers, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 4).

A reactor trip resulting from actuation of the ATWS Mitigation System Actuation Circuitry (AMSAC) logic that results in full insertion of control rods and diminishing neutron flux is considered a successful reactor trip. AMSAC automatically initiates auxiliary feedwater and a turbine trip under conditions indicative of an Anticipated Transient Without Scram (ATWS) event (ref. 5).

5% rated power is a minimum reading on the power range scale that indicates continued power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below 5%, plant response will be similar to that observed during a

normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than 5 % power (ref. 3, 4).

Escalation of this event to a SITE AREA EMERGENCY would be under EAL SS6.1 or SEC judgment.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(~~trip ~~[PWR] / scram [BWR])~~ that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor ~~(~~trip~~[PWR] / scram [BWR])~~). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control console~~s~~ (e.g., locally opening breakers). Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control console~~s~~". ~~Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action. [BWR]~~

The plant response to the failure of an automatic or manual reactor ~~(~~trip ~~[PWR] / scram [BWR])~~will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shut down the reactor is prolonged enough to cause a challenge to the core cooling ~~[PWR] / RPV water level [BWR]~~or RCS heat removal safety functions, the emergency classification level will escalate to a SITE AREA EMERGENCY via IC SS6~~5~~. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6~~5~~ or FS1, an ALERT declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an ALERT declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

CNP Basis Reference(s):

1. CNP Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. 1(2)-OHP04023-E-0 Reactor Trip or Safety Injection
3. 1(2)-OHP04023-F-0.1 Critical Safety Function Status Trees - Subcriticality
4. 1(2)-OHP-4023-FR-S-1 Response to Nuclear Power Generation/ATWS
5. UFSAR Section 3.3.3 Anticipated Transients Without Scram
6. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%

AND

All actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5%

AND EITHER:

- CSFST Core Cooling RED Path (F-0.2) conditions met
- CSFST Heat Sink RED Path (F-0.3) conditions met

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

This EAL addresses the following:

- Any automatic reactor trip signal followed by a manual trip that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed (EAL SA6.1), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

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.

Reactor shutdown achieved by use of FR-S.1 Response to Nuclear Power Generation/ATWS such as tripping the main turbine, locally opening reactor trip breakers, emergency boration or manually driving control rods are also credited as a successful manual trip provided reactor power can be reduced below 5% before indications of an extreme challenge to either core cooling or heat removal exist (ref. 1, 2).

5% rated power is a minimum reading on the power range scale that indicates continued power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below 5%, plant response will be similar to that observed during a

normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than 5 % power (ref. 1, 2).

Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED PATH conditions being met. Specifically, Core Cooling RED PATH conditions exist if either the five highest core exit TCs are reading greater than or equal to 1200°F or core exit TCs are reading greater than or equal to 757°F with RCS subcooling less than or equal 40°F and RVLIS level less than or equal to that specified based on the number of RCPs running (ref. 3).

Indication of inability to adequately remove heat from the RCS is manifested by CSFST Heat Sink RED PATH conditions being met (ref. 2). Specifically, Heat Sink RED PATH conditions exist if narrow range level in at least one steam generator is not greater than 13% (28% Adverse Containment Conditions) and total feedwater flow to the steam generators is less than or equal to 240,000 lbm/hr. (ref. 4).

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(~~trip ~~[PWR] / scram [BWR])~~ that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a SITE AREA EMERGENCY.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shut down the reactor. The inclusion of this IC and EAL ensures the timely declaration of a SITE AREA EMERGENCY in response to prolonged failure to shutdown the reactor.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Escalation of the emergency classification level would be via IC ~~AG1~~ RG1 or FG1.

CNP Basis Reference(s):

1. 1(2)-OHP04023-F-0.1 Critical Safety Function Status Trees - Subcriticality
2. 1(2)-OHP-4023-FR-S-1 Response to Nuclear Power Generation/ATWS
3. 1(2)-OHP04023-F-0.2 Critical Safety Function Status Trees – Core Cooling
4. 1(2)-OHP04023-F-0.3 Critical Safety Function Status Trees – Heat Sink
5. NEI 99-01 SS5

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

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

EAL:

SU7.1 Unusual Event

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

OR

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

OR

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

Table S-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Plant Page	X		
Plant Radios	X	X	
Plant Telephone	X	X	X
ENS Line		X	X
Commercial Telephone		X	X
Microwave Transmission		X	X

Mode Applicability:

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

Definition(s):

None

Basis:

Onsite/offsite communications include one or more of the systems listed in Table C-5 (ref. 1).

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

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

~~EAL #1~~The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

~~EAL #2~~The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are ~~(see Developer Notes)~~ the State and Berrien County EOCs~~.~~

~~EAL #3~~The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

CNP Basis Reference(s):

1. CNP Plant Emergency Plan Section F Emergency Communications
2. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 8 – Containment Failure

Initiating Condition: Failure to isolate containment or loss of containment pressure control.

EAL:

SU8.1 Unusual Event

Any penetration is **not** isolated within 15 min. of a VALID containment isolation signal

OR

Containment pressure > 2.8 psig with < one full train of containment depressurization equipment operating per design for ≥ 15 min. (Note 9)

(Note 1)

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

Note 9: One Containment Spray System train and one Containment Air Recirculation Fan comprise one full train of depressurization equipment.

Mode Applicability:

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

Definition(s):

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

Basis:

The containment isolation system provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the outside environment in the event of a design basis accident (ref. 1).

Containment pressure control is achieved through the Containment Spray System and the Containment Air Recirculation/Hydrogen Skimmer System. Failure of either of these systems may allow steam to build up within containment, and, unabated, this steam buildup may cause the internal containment pressure buildup to exceed the design pressure of 12 psig. Studies have shown that the containment can withstand pressures well above this value.

Both the recirculation fans and the containment spray pumps are actuated automatically (time delayed) following receipt of a HI or HI HI (Phase B) containment pressure signal, respectively. Since the HI HI containment pressure setpoint is less than or equal to 2.8 PSI, then greater than 2.8 PSI would be the containment pressure greater than the setpoint at which the equipment was supposed to have actuated per design. If these systems should fail to start automatically per design, a successful manual start within 15 minutes would preclude exceeding this Containment Potential Loss threshold. (ref. 2, 3, 4)

This ~~IC~~ EAL addresses a failure of one or more containment penetrations to automatically isolate (close) when required by an actuation signal. It also addresses an event that results in high containment pressure with a concurrent failure of containment pressure control systems. Absent challenges to another fission product barrier, either condition represents potential degradation of the level of safety of the plant.

For ~~EAL #1~~the first condition, the containment isolation signal must be generated as the result ~~on~~ of an off-normal/accident condition (e.g., a safety injection or high containment pressure); a failure resulting from testing or maintenance does not warrant classification. The determination of containment and penetration status – isolated or not isolated – should be made in accordance with the appropriate criteria contained in the plant AOPs and EOPs. This condition includes the failure of Containment Ventilation Isolation to actuate on a VALID signal. The 15-minute criterion is included to allow operators time to manually isolate the required penetrations, if possible.

~~EAL #2~~The second condition addresses a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment sprays or ~~ice condenser~~containment recirculation fans) are either lost or performing in a degraded manner.

This event would escalate to a SITE AREA EMERGENCY in accordance with IC FS1 if there were a concurrent loss or potential loss of either the Fuel Clad or RCS fission product barriers.

CNP Basis Reference(s):

1. UFSAR Section 5.4 Containment Isolation System
2. UFSAR Section 5.5.3 System Description
3. UFSAR Section 6.3 Containment Spray Systems
4. EC-0000052930 Unit 1 Return to Normal Operating Pressure and Temperature (NOP/NOT)
5. NEI 99-01 SU7

Category: S – System Malfunction

Subcategory: 9 – Hazardous Event Affecting Safety Systems

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

EAL:

SA9.1 Alert

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

AND EITHER:

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

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

Mode Applicability:

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

Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

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

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

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

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

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

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

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

VISIBLE DAMAGE - Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did **not** cause damage to a structure or any other equipment) is **not** visible damage.

Basis:

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

~~EAL 1.b.1~~The first condition addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

~~EAL 1.b.2~~The second condition addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC FS1 or ~~AS1~~RS1.

CNP Basis Reference(s):

1. NEI 99-01 SA9

Category F – Fission Product Barrier Degradation

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

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

A. Fuel Clad (FC): The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.
B. Reactor Coolant System (RCS): The RCS Barrier 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. Containment Barrier thresholds are used as criteria for escalation of the ECL from ALERT to a SITE AREA EMERGENCY or a GENERAL EMERGENCY.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 2). "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

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

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

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

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

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier.
- UNUSUAL EVENT ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.
- For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to

ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a SITE AREA EMERGENCY classification while a dose assessment may indicate that an EAL for GENERAL EMERGENCY IC RG1 has been exceeded.

- The fission product barrier thresholds specified within a scheme reflect plant-specific CNP design and operating characteristics.
- As used in this category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside the primary containment, an interfacing system, or outside of the primary containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered to be RCS leakage.
- At the SITE AREA EMERGENCY level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a GENERAL EMERGENCY declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the SEC would have more assurance that there was no immediate need to escalate to a GENERAL EMERGENCY.

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **Any** loss or **any** potential loss of either Fuel Clad or RCS

EAL:

FA1.1	**Alert**
Any loss or **any** potential loss of **EITHER** Fuel Clad **OR** RCS (Table F-1)	

Mode Applicability:

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

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the ALERT classification level, Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a SITE AREA EMERGENCY under EAL FS1.1.

CNP 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

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the SITE AREA EMERGENCY classification level, each barrier is weighted equally. A SITE AREA EMERGENCY is therefore appropriate for any combination of the following conditions:

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

At the SITE AREA EMERGENCY classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a GENERAL EMERGENCY is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a GENERAL EMERGENCY classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, the SEC would have greater assurance that escalation to a GENERAL EMERGENCY is less imminent.

CNP 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 third barrier

EAL:

FG1.1 General Emergency
Loss of **any** two barriers
AND
Loss or potential loss of third barrier (Table F-1)

Mode Applicability:

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

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (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

CNP Basis Reference(s):

1. NEI 99-01 FG1

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. RCS or SG Tube Leakage

B. Inadequate Heat removal

C. CNMT Radiation / RCS Activity

D. CNMT Integrity or Bypass

E. SEC Judgment

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

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A would be assigned "FC Loss A.1," the third Containment barrier Potential Loss in Category C would be assigned "CNMT P-Loss C.3," etc.

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

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

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

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

Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, and FA1.1 to determine the appropriate emergency classification.

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

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad (FC) Barrier		Reactor Coolant System (RCS) Barrier		Containment (CNMT) Barrier	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A RCS or SG Tube Leakage	None	None	1. An automatic or manual ECCS (SI) actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	1. Operation of a standby charging pump is required by **EITHER:** • UNISOLABLE RCS leakage • SG tube leakage 2. CSFST Integrity-RED Path (F-0.4) conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B Inadequate Heat Removal	1. CSFST Core Cooling-RED Path (F-0.2) conditions met	1. CSFST Core Cooling-ORANGE Path (F-0.2) conditions met 2. CSFST Heat Sink-RED Path (F-0.3)conditions met **AND** Heat sink is required	None	1. CSFST Heat Sink-RED Path (F-0.3) conditions met **AND** Heat sink is required	None	1. CSFST Core Cooling-RED Path (F-0.2) conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C CNMT Radiation / RCS Activity	1. Containment radiation > Table F-2 column "FC Loss" 2. Dose equivalent I-131 coolant activity > 300 μCi/cc	None	1. Containment radiation > Table F-2 column "RCS Loss"	None	None	1. Containment radiation > Table F-2 column "CNMT Potential Loss"
D CNMT Integrity or Bypass	None	None	None	None	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on SEC judgment • UNISOLABLE pathway from containment to the environment exists 2. Indications of RCS leakage outside of Containment	1. CSFST Containment-RED Path (F-0.5) conditions met 2. Containment hydrogen concentration ≥ 4% 3. Containment pressure > 2.8 psig with < one full train of depressurization equipment operating per design for ≥ 15 min. (Note 1, 9)
E SEC Judgment	1. **Any** condition in the opinion of the SEC that indicates loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the SEC that indicates potential loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the SEC that indicates loss of the RCS barrier	1. **Any** condition in the opinion of the SEC that indicates potential loss of the RCS barrier	1. **Any** condition in the opinion of the SEC that indicates loss of the Containment barrier	1. **Any** condition in the opinion of the SEC that indicates potential loss of the Containment barrier

Barrier: Fuel Clad

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

1. CSFST Core Cooling-RED Path (F-0.2) conditions met

Definition(s):

None

Basis:

Indication of continuing severe core cooling degradation is manifested by CSFST Core Cooling RED PATH conditions being met. Specifically, Core Cooling RED PATH conditions exist if either the five highest core exit TCs are reading greater than or equal to 1200°F or core exit TCs are reading greater than or equal to 757°F with RCS subcooling less than or equal 40°F and RVLIS level less than or equal to that specified based on the number of RCPs running (ref. 1).

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path indicates significant core exit superheating and core uncovery. The CSFSTs are normally monitored using the SPDS display on the Plant Process Computer (ref. 1, 2).

This reading indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

CNP Basis Reference(s):

1. 1(2)-OHP04023-F-0.2 Critical Safety Function Status Trees – Core Cooling
2. 1(2)-OHP-4023-FR-C.1 Response to Inadequate Core Cooling
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Core Cooling-ORANGE Path (F-0.2) conditions met

Definition(s):

None

Basis:

Indication of continuing significant core cooling degradation is manifested by CSFST Core Cooling ORANGE PATH conditions being met. Specifically, Core Cooling ORANGE PATH conditions exist if either the five highest core exit TCs are reading greater than or equal to 757°F with RCS subcooling less than or equal 40°F or RVLIS level less than or equal to that specified based on the number of RCPs running (ref. 1).

Critical Safety Function Status Tree (CSFST) Core Cooling-ORANGE path indicates subcooling has been lost and that some fuel clad damage may potentially occur. The CSFSTs are normally monitored using the SPDS display on the Plant Computer (ref. 1, 2).

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

CNP Basis Reference(s):

1. 1(2)-OHP04023-F-0.2 Critical Safety Function Status Trees – Core Cooling
2. 1(2)-OHP-4023-FR-C.2 Response to Degraded Core Cooling
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. CSFST Heat Sink-RED Path (F-0.3) conditions met

   AND

   Heat sink is required

Definition(s):

None

Basis:

In combination with RCS Potential Loss B.1, meeting this threshold results in a SITE AREA EMERGENCY.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED path indicates the ultimate heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

Heat Sink RED PATH conditions exist if narrow range level in all SGs is less than or equal to 13% and total feedwater flow to all SGs is less than or equal to 240,000 lbm/hr (ref. 1).

The CSFSTs are normally monitored using the SPDS display on the Plant Computer (ref. 2).

The phrase "and heat sink required" precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. For example, FR-H.1 is entered from CSFST Heat Sink-Red. Step 1 tells the operator to determine if heat sink is required by checking that RCS pressure is greater than any non-faulted SG pressure and RCS temperature is greater than 350°F. If these conditions exist, Heat Sink is required. Otherwise, the operator is to either return to the procedure and step in effect and place RHR in service for heat removal. For large LOCA events inside the Containment, the SGs are moot because heat removal through the containment heat removal systems takes place. Therefore, Heat Sink Red should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an ALERT classification. (ref. 2).

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the Fuel Clad Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

CNP Basis Reference(s):

1. 1(2)-OHP04023-F-0.3 Critical Safety Function Status Trees – Heat Sink
2. 1(2)-OHP-4023-FR-H.1 Response to Loss of Secondary Heat Sink
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.B

Barrier: Fuel Clad

Category: C. CNMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment radiation > Table F-2 column "FC Loss"

Table F-2 Containment Radiation – R/hr - VRA-1310 (2310) / 1410 (2410)			
Monitor	**FC Loss**	**RCS Loss**	**CNMT Potential Loss**
VRA-1310 (2310)	1,000	200	9,100
VRA-1410 (2410)	700	140	6,300

Definition(s):

None

Basis:

Containment radiation monitor readings greater than Table F-2 column "FC Loss" (ref. 1) indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading is derived assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 µCi/cc dose equivalent I-131 into the containment atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage (2 - 3% clad failure depending on core inventory and RCS volume). This value is higher than that specified for RCS barrier Loss C.1 (ref. 1, 2).

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors CHRM-VRA-1310/1410 (2310/2410).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300 µCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to ~~5~~3% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold ~~3.A~~C.1 since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the ~~emergency classification level~~ECL to a SITE AREA EMERGENCY.

CNP Basis Reference(s):

1. EP-CALC-CNP-1602, Containment Radiation EAL Threshold Values
2. EVAL-RD-99-11, Evaluation of Radiation Monitoring System Setpoints, Rev 0
3. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.A

Barrier: Fuel Clad

Category: C. CNMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. Dose equivalent I-131 coolant activity > 300 µCi/gm

Definition(s):

None

Basis:

This threshold indicates that RCS radioactivity concentration is greater than 300 µCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to ~~5~~3% fuel clad damage (ref. 1). Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

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

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

CNP Basis Reference(s):

1. EP-EALCALC-CNP-1602, Containment Radiation EAL Threshold Values
2. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.B

Barrier: Fuel Clad

Category: C. CNMT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CNMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CNMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. SEC Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the SEC that indicates loss of the Fuel Clad barrier

Definition(s):

None

Basis:

The SEC judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The SEC should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

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

CNP Basis Reference(s):

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

Barrier: Fuel Clad

Category: E. SEC Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the SEC that indicates potential loss of the Fuel Clad barrier

Basis:

The SEC judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The SEC should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

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

CNP Basis Reference(s):

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

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. An automatic or manual ECCS (SI) actuation required by **EITHER:**
   - UNISOLABLE RCS leakage
   - SG tube RUPTURE

Definition(s):

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

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

Basis:

ECCS (SI) actuation is caused by (ref. 1, 2):

- Pressurizer low pressure
- Steamline low pressure
- Lower Containment high pressure
- Steamline ΔP

This threshold is based on an UNISOLABLE RCS leak of sufficient size to require an automatic or manual actuation of the Emergency Core Cooling System (ECCS). This condition clearly represents a loss of the RCS Barrier.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED. If a RUPTURED steam generator is also FAULTED outside of containment, the declaration escalates to a SITE AREA EMERGENCY since the Containment Barrier Loss threshold ~~1.A~~A.1 will also be met.

CNP Basis Reference(s):

1. 1(2)-OHP-4023-E-0 Reactor Trip or Safety Injection
2. 1(2)-OHP-4023-E-3 Steam Generator Tube Rupture
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Loss 1.A

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

1. Operation of a standby charging pump is required by **EITHER:**
   - UNISOLABLE RCS leakage
   - SG tube leakage

Definition(s):

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

Basis:

This threshold is based on the inability to maintain liquid inventory within the RCS by normal operation of the Chemical and Volume Control System (CVCS). The CVCS includes three charging pumps: one positive displacement pump with a flow capacity of 150 gpm, and two centrifugal charging pumps each with a flow capacity of 150 gpm (ref. 1). A second charging pump being required is indicative of a substantial RCS leak.

This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ECCS (SI) actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs that a standby charging (makeup) pump be placed in service to restore and maintain pressurizer level.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

If a leaking steam generator is also FAULTED outside of containment, the declaration escalates to a SITE AREA EMERGENCY since the Containment Barrier Loss threshold 1.A will also be met.

CNP Basis Reference(s):

1. UFSAR Table 9.2-2 Chemical and Volume Control System Design Parameters
2. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.A

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

2. CSFST Integrity-RED Path (F-0.4) conditions met

Definition(s):

None

Basis:

The "Potential Loss" threshold is defined by the CSFST Reactor Coolant Integrity - RED path. CSFST RCS Integrity - Red Path plant conditions and associated PTS Limit Curve A indicates an extreme challenge to the safety function when plant parameters are to the left of the limit curve following excessive RCS cooldown under pressure (ref. 1, 2).

This condition indicates an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized).

CNP Basis Reference(s):

1. 1(2)-OHP-4023-F-0.4 Critical Safety Function Status Trees Figure F-0.4-1 Integrity Operational Limits
2. 1(2)-OHP-4023-FR-P.1 Response to Imminent Pressurized Thermal Shock Condition
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.B

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Heat Sink-RED Path (F-0.3) conditions met

   AND

   Heat sink is required

Definition(s):

None

Basis:

In combination with FC Potential Loss B.2, meeting this threshold results in a SITE AREA EMERGENCY.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED path indicates the ultimate heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

Heat Sink RED PATH conditions exist if narrow range level in all SGs is less than or equal to 13% and total feedwater flow to all SGs is less than or equal to 240,000 lbm/hr (ref. 1).

The CSFSTs are normally monitored using the SPDS display on the Plant Computer (ref. 2).

The phrase "and heat sink required" precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. For example, FR-H.1 is entered from CSFST Heat Sink-Red. Step 1 tells the operator to determine if heat sink is required by checking that RCS pressure is greater than any non-faulted SG pressure and RCS temperature is greater than 350°F. If these conditions exist, Heat Sink is required. Otherwise, the operator is to either return to the procedure and step in effect and place RHR in service for heat removal. For large LOCA events inside the Containment, the SGs are moot because heat removal through the containment heat removal systems takes place. Therefore, Heat Sink Red should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an ALERT classification. (ref. 2).

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the RCS Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

Meeting this threshold results in a SITE AREA EMERGENCY because this threshold is identical to Fuel Clad Barrier Potential Loss threshold ~~2.~~B.2; both will be met. This condition warrants a SITE AREA EMERGENCY declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase RCS pressure to the point where mass will be lost from the system.

CNP Basis Reference(s):

1. 1(2)-OHP04023-F-0.3 Critical Safety Function Status Trees – Heat Sink
2. 1(2)-OHP-4023-FR-H.1 Response to Loss of Secondary Heat Sink
3. NEI 99-01 Inadequate Heat Removal RCS Loss 2.B

Barrier: Reactor Coolant System

Category: C. CNMT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment radiation > Table F-2 column "RCS Loss"

Table F-2 Containment Radiation – R/hr - VRA-1310 (2310) / 1410 (2410)			
Monitor	**FC Loss**	**RCS Loss**	**CNMT Potential Loss**
VRA-1310 (2310)	1,000	200	9,100
VRA-1410 (2410)	700	140	6,300

Definition(s):

N/A

Basis:

Containment radiation monitor readings greater than Table F-2 column "RCS Loss" (ref. 1, 2) indicate the release of reactor coolant to the containment. The readings assume 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 primary coolant.

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors CHRM-VRA-1310/1410 (2310/2410).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold ~~3.A~~C.1 since it indicates a loss of the RCS Barrier only.

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

CNP Basis Reference(s):

1. EP-CALC-CNP-1602, Containment Radiation EAL Threshold Values
2. EVAL-RD-99-11, Evaluation of Radiation Monitoring System Setpoints, Rev 0
3. NEI 99-01 CMT Radiation / RCS Activity RCS Loss 3.A

Barrier: Reactor Coolant System

Category: B. CNMT Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CNMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: D. CNMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. SEC Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the SEC that indicates loss of the RCS barrier

Definition(s):

None

Basis:

The SEC judgment threshold addresses any other factors relevant to determining if the RCS barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The SEC should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

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

CNP Basis Reference(s):

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

Barrier: Reactor Coolant System

Category: E. SEC Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the SEC that indicates potential loss of the RCS barrier

Definition(s):

None

Basis:

The SEC judgment threshold addresses any other factors relevant to determining if the RCS barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The SEC should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

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

CNP Basis Reference(s):

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

Barrier: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. A leaking or RUPTURED SG is FAULTED outside of containment

Definition(s):

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

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

Basis:

This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for RCS Barrier Potential Loss ~~1.~~A.1 and Loss ~~1.~~A.1, respectively. This condition represents a bypass of the containment barrier.

FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably (~~{~~part of the FAULTED definition)~~}~~ and the FAULTED steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition, the steam generator is still considered FAULTED for emergency classification purposes.

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU4 for the fuel clad barrier (i.e., RCS activity values) and IC SU5 for the RCS barrier (i.e., RCS leak rate values).

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, glad seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category ~~A~~ R ICs.

The ~~emergency classification level~~ECLs resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

	Affected SG is FAULTED Outside of Containment?	
P-to-S Leak Rate	**Yes**	**No**
Less than or equal to 25 gpm	No classification	No classification
Greater than 25 gpm	UNUSUAL EVENT per ~~SU4~~SU5.1	UNUSUAL EVENT per ~~SU4~~SU5.1
Requires operation of a standby charging (makeup) pump (*RCS Barrier Potential Loss*)	SITE AREA EMERGENCY per FS1.1	ALERT per FA1.1
Requires an automatic or manual ECCS (SI) actuation (*RCS Barrier Loss*)	SITE AREA EMERGENCY per FS1.1	ALERT per FA1.1

There is no Potential Loss threshold associated with RCS or SG Tube Leakage.

CNP Basis Reference(s):

1. NEI 99-01 RCS or SG Tube Leakage Containment Loss 1.A

Barrier: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: B. Inadequate heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: B. Inadequate heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Core Cooling-RED Path (F-0.2) conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)

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

Definition(s):

None

Basis:

Indication of continuing severe core cooling degradation is manifested by CSFST Core Cooling RED PATH conditions being met. Specifically, Core Cooling RED PATH conditions exist if either the five highest core exit TCs are reading greater than or equal to 1200°F or core exit TCs are reading greater than or equal to 757°F with RCS subcooling less than or equal 40°F and RVLIS level less than or equal to that specified based on the number of RCPs running (ref. 1).

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path indicates significant core exit superheating and core uncovery. The CSFSTs are normally monitored using the SPDS display on the Plant Computer (ref. 1).

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 (ref. 1, 2, 3).

A direct correlation to status trees can be made if the effectiveness of the restoration procedures is also evaluated. If core exit thermocouple (TC) readings are greater than 1,200°F (ref. 1), Fuel Clad barrier is also lost.

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

CNP Basis Reference(s):

1. 1(2)-OHP04023-F-0.2 Critical Safety Function Status Trees – Core Cooling
2. 1(2)-OHP-4023-FR-C.1 Response to Inadequate Core Cooling
3. 1(2)-OHP04023-FR-C.2 Response to Degraded Core Cooling
4. NEI 99-01 Inadequate Heat Removal Containment Potential Loss 2.A

Barrier: Containment

Category: C. CNMT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: C. CNMT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

1. Containment radiation > Table F-2 column "CNMT Potential Loss"

Table F-2 Containment Radiation – R/hr - VRA-1310 (2310) / 1410 (2410)			
Monitor	**FC Loss**	**RCS Loss**	**CNMT Potential Loss**
VRA-1310 (2310)	1,000	200	9,100
VRA-1410 (2410)	700	140	6,300

Definition(s):

None

Basis:

Containment radiation monitor readings greater than Table F-2 column "CNMT Potential Loss" (ref. 1, 2) indicate significant fuel damage (20% clad damage) well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier.

The readings are higher than that specified for Fuel Clad barrier Loss C.1 and RCS barrier Loss C.1. 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.

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors CHRM-VRA-1310/1410 (2310/2410).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the ~~emergency classification level~~ECL to a GENERAL EMERGENCY.

CNP Basis Reference(s):

1. EP-CALC-CNP-1602, Containment Radiation EAL Threshold Values
2. EVAL-RD-99-11, Evaluation of Radiation Monitoring System Setpoints, Rev 0
3. NEI 99-01 CMT Radiation / RCS Activity Containment Potential Loss 3.A

Barrier: Containment

Category: D. CNMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. Containment isolation is required

   AND EITHER:

   - Containment integrity has been lost based on SEC judgment
   - UNISOLABLE pathway from containment to the environment exists

Definition(s):

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

Basis:

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold ~~1.~~A.1.

These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both bulleted thresholds~~4.A.1 and 4.A.2~~.

~~4.A.1~~First Threshold – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the Site Emergency ~~Director~~ Coordinator will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure ~~9-F-4~~1. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

Another example would be a loss or potential loss of the RCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category ~~A~~R ICs.

~~4.A.2~~Second Threshold – Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment atmosphere to the environment. As used here, the term "environment" includes the atmosphere of a room or area, outside the containment, that may, in turn, communicate with the outside-the-plant atmosphere (e.g., through discharge of a ventilation system or atmospheric leakage). Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.

Refer to the top piping run of Figure ~~9-F-4~~1. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Leakage between two interfacing liquid systems, by itself, does not meet this threshold.

Refer to the bottom piping run of Figure ~~9-F-4~~1. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump developed a leak that allowed steam/water to enter the Auxiliary Building, then second threshold ~~4.B~~ would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause the first threshold ~~4.A.1~~ to be met as well.

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

CNP Basis Reference(s):

1. NEI 99-01 CMT Integrity or Bypass Containment Loss 4.A

Barrier: Containment

Category: D. CNMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

2. Indications of RCS leakage outside of containment

Definition(s):

None

Basis:

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold A.1.

To ensure proper escalation of the emergency classification, the RCS leakage outside of containment must be related to the mass loss that is causing the RCS Loss and/or Potential Loss threshold ~~1.~~A.1 to be met.

ECA-1.2 LOCA Outside Containment (ref. 1) provides instructions to identify and isolate a LOCA outside of the containment. Potential RCS leak pathways outside containment include (ref. 1, 2):

- Residual Heat Removal
- Safety Injection
- Chemical & Volume Control
- RCP seals

Containment sump, temperature, pressure and/or radiation levels will increase if reactor coolant mass is leaking into the containment. If these parameters have not increased, then the reactor coolant mass may be leaking outside of containment (i.e., a containment bypass sequence). Increases in sump, temperature, pressure, flow and/or radiation level readings outside of the containment may indicate that the RCS mass is being lost outside of containment.

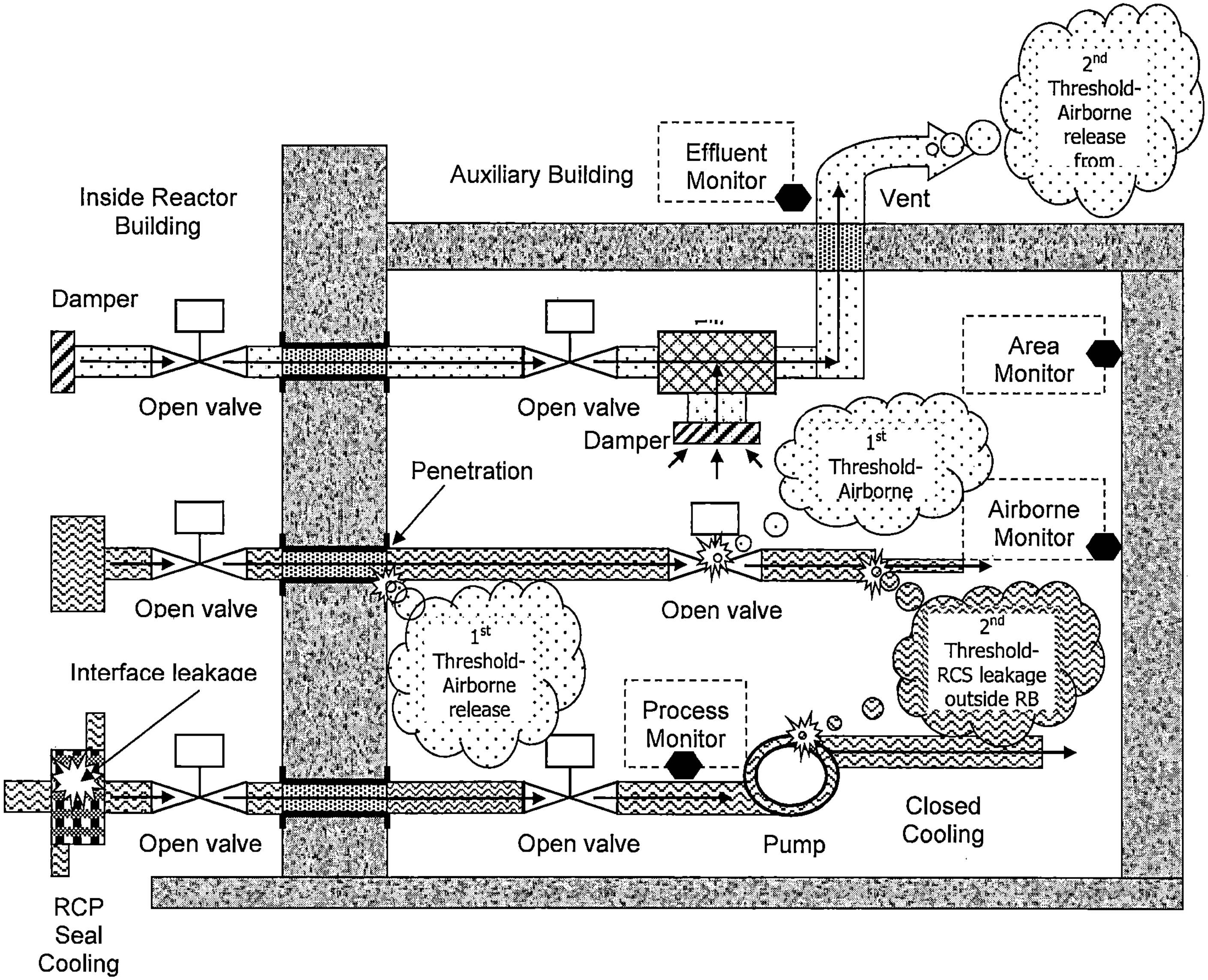
Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of RCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside of containment may not increase significantly; however, other unexpected changes in sump levels, area temperatures or pressures, flow rates, etc. should be sufficient to determine if RCS mass is being lost outside of the containment.

Refer to the middle piping run of Figure ~~9-F-4~~1. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause threshold ~~4.A~~D.1 to be met as well.

CNP Basis Reference(s):

1. 1(2)-OHP-4023-ECA-1.2 LOCA Outside Containment
2. 1(2)-OHP-4023-E-1 Loss of Reactor or Secondary Coolant
3. NEI 99-01 CMT Integrity or Bypass Containment Loss

Figure 1: Containment Integrity or Bypass Examples



Barrier: Containment

Category: D. CNMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

1. CSFST Containment-RED Path (F-0.5) conditions met

Definition(s):

None

Basis:

Critical Safety Function Status Tree (CSFST) Containment-RED path is entered if containment pressure is greater than or equal to 12 psig and represents an extreme challenge to safety function. The CSFSTs are normally monitored using the SPDS display on the Plant Computer (ref. 1, 2).

12 psig is the containment design pressure (ref. 3) and is the pressure used to define CSFST Containment Red Path conditions.

If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the RCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a SITE AREA EMERGENCY and GENERAL EMERGENCY since there is now a potential to lose the third barrier.

CNP Basis Reference(s):

1. 1(2)-OHP-4032-F00.5 Critical Safety Function Status Trees Containment
2. 1(2)-OHP-4023-FR-Z.1 Response to High Containment Pressure
3. UFSAR Section 5.2.2.2 Design Load Criteria
4. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.A

Barrier: Containment

Category: D. CNMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

2. Containment hydrogen concentration ≥ 4%

Definition(s):

None

Basis:

Following a design basis accident, hydrogen gas may be generated inside the containment by reactions such as zirconium metal with water, corrosion of materials of construction and radiolysis of aqueous solution in the core and sump. The lower limit of combustion of hydrogen in air is approximately 4%.

CNP is equipped with a Post-Accident Hydrogen Monitoring System (PACHMS) which serves to measure combustible gas concentrations in the containment. The PACHMS is comprised of two sampling-analyzing-control trains (ref. 1).

To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers must have occurred. With the Potential Loss of the containment barrier, the threshold hydrogen concentration, therefore, will likely warrant declaration of a GENERAL EMERGENCY.

Two Containment hydrogen monitors with duel ranges of 0% to 10% and 0% to 30% provide indication locally and in the Control Room (ref. 1, 2).

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

CNP Basis Reference(s):

1. UFSAR Section 7.8.2 Post-Accident Hydrogen Monitoring
2. 12-THP-6020-PAS-003, Post Accident Containment Hydrogen Monitoring System Operation
3. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.B

Barrier: Containment

Category: D. CNMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

3. Containment pressure > 2.8 psig with < one full train of containment depressurization equipment operating per design for ≥ 15 min. (Notes 1, 9)

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

Note 9: One Containment Spray System train and one Containment Air Recirculation Fan comprise one full train of depressurization equipment.

Definition(s):

None

Basis:

Containment pressure control is achieved through the Containment Spray System and the Containment Air Recirculation/Hydrogen Skimmer System. Failure of either of these systems may allow steam to build up within containment, and, unabated, this steam buildup may cause the internal containment pressure buildup to exceed the design pressure of 12 psig. Studies have shown that the containment can withstand pressures well above this value.

Both the recirculation fans and the containment spray pumps are actuated automatically (delayed) following receipt of a HI or HI HI containment pressure signal, respectively. Since the HI HI containment pressure setpoint is less than or equal to 2.8 PSI, then greater than 2.8 PSI would be the containment pressure greater than the setpoint at which the equipment was supposed to have actuated. If these systems should fail to start automatically per design, a successful manual start within 15 minutes would preclude exceeding this Containment Potential Loss threshold. (ref. 1, 2, 3).

This threshold describes a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems (e.g., containment sprays, ~~ice condenser~~containment recirculation fans, etc., but not including containment venting strategies) are either lost or performing in a degraded manner.

CNP Basis Reference(s):

1. UFSAR Section 5.5.3 System Description
2. UFSAR Section 6.3 Containment Spray Systems
3. EC-0000052930 Unit 1 Return to Normal Operating Pressure and Temperature (NOP/NOT)
4. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.C

Barrier: Containment

Category: E. SEC Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the SEC that indicates loss of the Containment barrier

Definition(s):

None

Basis:

The SEC judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The SEC should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

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

CNP Basis Reference(s):

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

Barrier: Containment

Category: E. SEC Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the SEC that indicates potential loss of the Containment barrier

Definition(s):

None

Basis:

The SEC judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The SEC should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

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

CNP Basis Reference(s):

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

Background

NEI 99-01 Revision 6 ICs AA3 and HA5 prescribe declaration of an ALERT based on impeded access to rooms or areas (due to either area radiation levels or hazardous gas concentrations) where equipment necessary for normal plant operations, cooldown or shutdown is located. These areas are intended to be plant operating mode dependent. Specifically the Developers Notes For AA3 and HA5 states:

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

The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).

Further, as specified in IC HA5:

The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.

CNP Table R-2 and H-2 Bases

A review of station operating procedures identified the following mode dependent in-plant actions and associated rooms or areas that are required for normal plant operation, cooldown or shutdown:

In-Plant Actions	Safe Shutdown Area	Modes
Condensate, Feedwater & Main / Feed Pump Turbine Operating Procedures	Turbine Building All Levels	1, 2, 3
Reactor Coolant Sampling	Auxiliary 587' & Aux 609'	1, 2, 3, 4, 5
Steam Generator Blowdown System Operation	Auxiliary 587', 591', & 633'	1, 2, 3, 4
Operation of Screen Wash and Traveling Screens	Screenhouse	1, 2, 3, 4, 5
Auxiliary Feedwater Pump Operations	Turbine Building 591'	1, 2, 3, 4, 5
Operation of the Residual Heat Removal System	Auxiliary 573' & Aux 609'	4, 5
ECCS Breaker Alignments	Auxiliary 587', 609' & 633'	5

Table R-2 & H-2 Results

Table R-2 & H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Auxiliary Building 573'	4, 5
Auxiliary Building 587' (including D/G room)	1, 2, 3, 4, 5
Auxiliary Building 591'	1, 2, 3, 4
Auxiliary Building 609' (including 4kV room)	1, 2, 3, 4, 5
Auxiliary Building 633'	1, 2, 3, 4
Turbine Building (All Levels)	1, 2, 3
Turbine Building 591'	4, 5
Screenhouse	1, 2, 3, 4, 5

Plant Operating Procedures Reviewed

Mode	Procedure	Operation
1	OHP-4021-011-001	At-Power Operation Including Load Swings
1-3	OHP-4021-001-003	Power Reduction
1-5	THP-6020-CHM-121	Reactor Coolant Sampling
1-4	OHP-4021-025-001	Steam Generator Blowdown
3-5	OHP-4021-001-004	Plant Cooldown from Hot Standby to Cold Shutdown
4-5	OHP-4021-017-002	Placing in Service the Residual Heat Removal System
1-5	OHP-4021-056-001	Auxiliary Feed Pump Operation
1-5	OHP-4021-057-005	Operation of Screen Wash and Traveling Screens
1-5	OHP-4021-019-001	Operation of the Essential Service Water System
1-5	OHP-4021-018-002	Placing in Service and Operating the Spent Fuel Pit Cooling and Cleanup System

Enclosure 4 to AEP-NRC-2017-02

DONALD C. COOK NUCLEAR PLANT EMERGENCY PLAN
EAL TECHNICAL BASIS MANUAL CLEAN PAGES

AEP: D.C. Cook
EAL Technical Basis Manual

Revision 0

TABLE OF CONTENTS

1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for D. C. Cook Nuclear Plant (CNP). Decision-makers responsible for implementation of PMP-2080-EPP-101 Emergency Classification, may use this document as a technical reference in support of EAL interpretation. This information may assist the SITE EMERGENCY COORDINATOR (SEC) in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training and for explaining event classifications to off-site officials.

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

Because the information in a basis document can affect emergency classification decision-making (e.g., the Emergency Coordinator refers to it during an event), the NRC staff expects that changes to the basis document will be evaluated in accordance with the provisions of 10 CFR 50.54(q). Additionally, changes to plant AOPs and EOPs that may impact EAL bases shall be evaluated in accordance with the provisions of 10 CFR 50.54(q).

2.0 DISCUSSION

2.1 Background

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

In 1992, the NRC endorsed NUMARC/NESP-007 “Methodology for Development of Emergency Action Levels” as an alternative to NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” EAL guidance.

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

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

Subsequently, Revision 6 of NEI 99-01 "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 (ADAMS Accession Number ML12326A805) (ref. 4.1.1) was issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, CNP conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

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

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

The primary fission product barriers are:

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

B. Reactor Coolant System (RCS): The RCS Barrier 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. Containment Barrier thresholds are used as criteria for escalation of the Emergency Classification Level (ECL) from ALERT to a SITE AREA EMERGENCY or a GENERAL EMERGENCY.

2.3 Fission Product Barrier Classification Criteria

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

Alert:

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

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

2.4 EAL Organization

The CNP EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
    - EALs applicable under any 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 group, assignment of EALs to categories and subcategories:

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

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
R – Abnormal Rad Levels / Rad Effluent	1 – Radiological Effluent 2 – Irradiated Fuel Event 3 – Area Radiation Levels
H – Hazards and Other Conditions Affecting Plant Safety	1 – Security 2 – Seismic Event 3 – Natural or Technological Hazard 4 – Fire 5 – Hazardous Gas 6 – Control Room Evacuation 7 – Emergency Coordinator Judgment
E – ISFSI	1 – Confinement Boundary
Hot Conditions:	
S – System Malfunction	1 – Loss of Emergency AC Power 2 – Loss of Vital DC Power 3 – Loss of Control Room Indications 4 – RCS Activity 5 – RCS Leakage 6 – RPS Failure 7 – Loss of Communications 8 – Containment Failure 9 – Hazardous Event Affecting Safety Systems
F – Fission Product Barrier Degradation	None
Cold Conditions:	
C – Cold Shutdown / Refueling System Malfunction	1 – RCS Level 2 – Loss of Emergency AC Power 3 – RCS Temperature 4 – Loss of Vital DC Power 5 – Loss of Communications 6 – Hazardous Event Affecting Safety Systems

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

2.5 Technical Bases Information

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

Category Letter & Title

Subcategory Number & Title

Initiating Condition (IC)

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

EAL Identifier (enclosed in rectangle)

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

1. First character (letter): Corresponds to the EAL category as described above (R, C, H, S, E 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).

Classification (enclosed in rectangle):

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

EAL (enclosed in rectangle)

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

Mode Applicability

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

Definitions:

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

Basis:

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

CNP Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.6 Operating Mode Applicability (ref. 4.1.7)

1 Power Operation

$K_{eff} \geq 0.99$ and reactor thermal power > 5%

2 Startup

$K_{eff} \geq 0.99$ and reactor thermal power ≤ 5%

3 Hot Standby

$K_{eff} < 0.99$ and average coolant temperature ≥ 350°F

4 Hot Shutdown

$K_{eff} < 0.99$ and average coolant temperature 350°F > T_{avg} > 200

5 Cold Shutdown

$K_{eff} < 0.99$ and average coolant temperature ≤ 200°F

6 Refueling

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

D Defueled

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

The mode in effect at the time that an event or condition occurred, and prior to any plant or operator response, is the mode that determines whether or not an IC is applicable. If an event or condition occurs, and results in a mode change before the emergency is declared, the emergency classification level is still based on the mode that existed at the time that the event or condition was initiated (and not when it was declared). Once a different mode is reached, any new event or condition, not related to the original event or condition, requiring emergency classification should be evaluated against the ICs and EALs applicable to the operating mode at the time of the new event or condition. For events that occur in Cold Shutdown or Refueling, escalation is via EALs that are applicable in the Cold Shutdown or Refueling modes, even if Hot Shutdown (or a higher mode) is entered during the subsequent plant response. In particular, the fission product barrier EALs are applicable only to events that initiate in the Hot Shutdown mode or higher.

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Emergency Coordinator must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes, and the informing basis information. In the Recognition Category F matrices, EALs are based on loss or potential loss of Fission Product Barrier Thresholds.

3.1.1 Classification Timeliness

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, "Interim Staff Guidance, Emergency Planning for Nuclear Power Plants" (ref. 4.1.10).

When assessing an EAL that specifies a time duration for the off-normal condition, the "clock" for the EAL time duration runs concurrently with the emergency classification process "clock."

3.1.2 Valid Indications

All emergency classification assessments shall be based upon valid indications, reports or conditions. A valid indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy. For example, verification could be accomplished through an instrument channel check, response on related or redundant indicators, or direct observation by plant personnel. The validation of indications should be completed in a manner that supports timely emergency declaration.

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

3.1.3 Imminent Conditions

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

3.1.4 Planned vs. Unplanned Events

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

3.1.5 Classification Based on Analysis

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

3.1.6 SITE EMERGENCY COORDINATOR (SEC) Judgment

While the EALs have been developed to address a full spectrum of possible events and conditions which may warrant emergency classification, a provision for classification based on operator/management experience and judgment is still necessary. The NEI 99-01 EAL scheme provides the SEC with the ability to classify events and conditions based upon judgment using EALs that are consistent with the Emergency Classification Level (ECL) definitions (refer to Category H). The SEC will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated in the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

3.2 Classification Methodology

To make an emergency classification, the user will compare an event or condition (i.e., the relevant plant indications and reports) to an EAL(s) and determine if the EAL has been met or exceeded. The evaluation of an EAL must be consistent with the related Operating Mode Applicability and Notes. If an EAL has been met or exceeded, the associated IC is likewise met, the emergency classification process "clock" starts, and the ECL must be declared in accordance with plant procedures no later than fifteen minutes after the process "clock" started.

When assessing an EAL that specifies a time duration for the off-normal condition, the "clock" for the EAL time duration runs concurrently with the emergency classification process "clock." For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01 (ref. 4.1.10).

3.2.1 Classification of Multiple Events and Conditions

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

- If an ALERT EAL and a SITE AREA EMERGENCY EAL are met, whether at one unit or at two different units, a SITE AREA EMERGENCY should be declared.

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

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

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

3.2.2 Consideration of Mode Changes During Classification

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

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

3.2.3 Classification of Imminent Conditions

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

3.2.4 Emergency Classification Level Upgrading and Downgrading

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

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

3.2.5 Classification of Short-Lived Events

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

3.2.6 Classification of Transient Conditions

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

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

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

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

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

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

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

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022, Event Report Guidelines 10 CFR 50.72 and 50.73, (ref. 4.1.3) is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR § 50.72 (ref. 4.1.4) within one hour of the discovery of the undeclared event or condition. The licensee should also notify appropriate State and local agencies in accordance with the agreed upon arrangements.

3.2.8 Retraction of an Emergency Declaration

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

4.0 REFERENCES

4.1 Developmental

4.1.1 NEI 99-01 Revision 6, Methodology for the Development of Emergency Action Levels for Non-Passive Reactors, ADAMS Accession Number ML12326A805

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

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

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

4.1.5 10 § CFR 50.73 License Event Report System

4.1.6 CNP UFSAR Figure 1.3-1 Plot Plan

4.1.7 Technical Specifications Table 1.1-1 Modes

4.1.8 PMP-4100-SDR-001 Plant Shutdown Safety and Risk Management

4.1.9 PMP-2010-PRC-001 Procedure Writing

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

4.1.11 CNP Emergency Plan

4.2 Implementing

4.2.1 PMP-2080-EPP-101 Emergency Classification

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

4.2.3 CNP EAL Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

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

ALERT

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

CONTAINMENT CLOSURE

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

As applied to CNP, Containment Closure is established when the requirements of PMP-4100-SDR-001 are met (ref. 4.1.8).

CONFINEMENT BOUNDARY

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As applied to the CNP ISFSI, the CONFINEMENT BOUNDARY is defined to be the Multi-Purpose Canister (MPC).

EMERGENCY ACTION LEVEL

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

EMERGENCY CLASSIFICATION LEVEL

One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are: UNUSUAL EVENT (UE), ALERT, SITE AREA EMERGENCY (SAE) and GENERAL EMERGENCY (GE).

EPA PAGs

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

EXPLOSION

A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

FAULTED

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

FIRE

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

FISSION PRODUCT BARRIER THRESHOLD

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

FLOODING

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

GENERAL EMERGENCY

Events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or hostile actions that result in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

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 CNP 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 CNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

HOSTILE FORCE

One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT

The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

IMPEDE(D)

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

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

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

Initiating Condition

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

INTACT (RCS)

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

MAINTAIN

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

OWNER CONTROLLED AREA

The property associated with the station and owned by the company. Access is normally limited to persons entering for official business.

PROJECTILE

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

PROTECTED AREA

The area encompassed by physical barriers to control access to the plant and to the ISFSI. (ref. 4.1.6).

RCS INTACT

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

Reduced Inventory

Operating condition when fuel is in the reactor vessel and Reactor Coolant System level is lower than 3 feet (or more) below the Reactor Vessel flange (ref. 4.1.8).

REFUELING PATHWAY

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

RUPTURED

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

RESTORE

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

SAFETY SYSTEM

A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

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

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

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

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

SECURITY CONDITION

Any security event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A security condition does not involve a hostile action.

SITE AREA EMERGENCY

Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or hostile actions that result 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 Guidelines exposure levels beyond the site boundary.

SITE EMERGENCY COORDINATOR (SEC)

The individual who has the responsibility for event classification, event notification, and approval of protective action recommendations to offsite organizations. It is recognized that during an emergency event this responsibility can be formally turned over from the Shift Manager, to a Site Emergency Coordinator located in the TSC, or to the Emergency Director located in the EOF as the response facilities become activated during an emergency event.

UNISOLABLE

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

UNPLANNED

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

UNUSUAL EVENT

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

VALID

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

VISIBLE DAMAGE

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

5.2 Abbreviations/Acronyms

°F .... Degrees Fahrenheit
AC .... Alternating Current
ATWS .... Anticipated Transient Without Scram
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
CHRM .... Containment High Range Monitor
CNMT .... Containment
CNP .... D. C. Cook Nuclear Plant
CSFST .... Critical Safety Function Status Tree
D .... Defueled
DBA .... Design Basis Accident
DBT .... Design Basis Threat
DC .... Direct Current
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
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
FPB .... Fission Product Barrier
FSAR .... Final Safety Analysis Report
GE .... General Emergency
Hrs .... Hours
IC .... Initiating Condition
IPEEE .... Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI .... Independent Spent Fuel Storage Installation
K_{eff} .... Effective Neutron Multiplication Factor
LCO .... Limiting Condition of Operation
LER .... Licensee Event Report
LOCA .... Loss of Coolant Accident
LWR .... Light Water Reactor
MPC .... Maximum Permissible Concentration/Multi-Purpose Canister

mR, mRem, mrem, mREM .... milli-Roentgen Equivalent Man
MSL .... Main Steam Line
NEI .... Nuclear Energy Institute
NESP .... National Environmental Studies Project
NPP .... Nuclear Power Plant
NRC .... Nuclear Regulatory Commission
NSSS .... Nuclear Steam Supply System
NORAD .... North American Aerospace Defense Command
(NO)UE .... Notification of Unusual Event
NUMARC .... Nuclear Utility Management and Resource Council
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM .... Off-site Dose Calculation Manual
ORO .... Offsite Response Organization
PA .... Protected Area
PAG .... Protective Action Guideline
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR .... Pressurized Water Reactor
PSIG .... Pounds per Square Inch Gauge
R .... Roentgen
RCC .... Reactor Control Console
RCS .... Reactor Coolant System
Rem, rem, REM .... Roentgen Equivalent Man
RETS .... Radiological Effluent Technical Specifications
RPS .... Reactor Protection System
R(P)V .... Reactor (Pressure) Vessel
RVLIS .... Reactor Vessel Level Indicating System
S/D .... Shutdown
SAR .... Safety Analysis Report
SBO .... Station Blackout
SCBA .... Self-Contained Breathing Apparatus
SEC .... Site Emergency Coordinator
SG .... Steam Generator
SI .... Safety Injection
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
TEDE .... Total Effective Dose Equivalent
TOAF .... Top of Active Fuel
TSC .... Technical Support Center
WOG .... Westinghouse Owners Group

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

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

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

CNP	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
CU1.2	CU1	2
CU2.1	CU2	1
CU3.1	CU3	1
CU3.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1, 2, 3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA3.1	CA3	1, 2
CA6.1	CA6	1
CS1.1	CS1	3
CG1.1	CG1	2
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1, 2 3
HU2.1	HU2	1
HU3.1	HU3	1
HU3.2	HU3	2
HU3.3	HU3	3
HU3.4	HU3	4
HU4.1	HU4	1
HU4.2	HU4	2
HU4.3	HU4	3

CNP	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
HU4.4	HU4	4
HU7.1	HU7	1
HA1.1	HA1	1, 2
HA5.1	HA5	1
HA6.1	HA6	1
HA7.1	HA7	1
HS1.1	HS1	1
HS6.1	HS6	1
HS7.1	HS7	1
HG7.1	HG7	1
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	2
SU5.1	SU4	1, 2, 3
SU6.1	SU5	1
SU6.2	SU5	2
SU7.1	SU6	1, 2, 3
SU8.1	SU7	1, 2
SA1.1	SA1	1
SA3.1	SA2	1
SA6.1	SA5	1
SA9.1	SA9	1
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1

CNP	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
SG1.1	SG1	1
SG2.1	SG8	1
EU1.1	E-HU1	1

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Fission Product Barrier Matrix and Basis

7.3 Safe Operation & Shutdown Rooms/Areas Tables R-2 & H-2 Bases

Category R – Abnormal Rad Release / Rad Effluent

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

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

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

Events of this category pertain to the following subcategories:

1. Radiological Effluent

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

2. Irradiated Fuel Event

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

3. Area Radiation Levels

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

ATTACHMENT 1
EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.1 Unusual Event

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

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

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

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

Table R-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	ALERT	UE
Gaseous	Unit Vent Noble Gas	VRS-1500 (2500)	3.3E+00 μCi/cc	3.3E-01 μCi/cc	3.3E-02 μCi/cc	4.2E-03 μCi/cc
Gaseous	Gland Seal Leakoff	SRA-1800 (2800)	1.6E+02 μCi/cc	1.6E+01 μCi/cc	1.6E+00 μCi/cc	1.4E-01 μCi/cc
Gaseous	Steam Jet Air Ejector	SRA-1900 (2900)	1.5E+04 μCi/cc	1.5E+03 μCi/cc	1.5E+02 μCi/cc	1.3E+01 μCi/cc
Liquid	Radwaste Effluent	RRS-1001	—	—	—	4.6E+04 cpm
Liquid	SG Blowdown	R-19	—	—	—	1.7E+03 cpm
Liquid	SG Blowdown	DRS-3100/4100	—	—	—	1.2E+04 cpm
Liquid	SG Blowdown Treatment	R-24	—	—	—	2.9E+04 cpm
Liquid	SG Blowdown Treatment	DRS-3200/4200	—	—	—	1.2E+05 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

The column "UE" gaseous and liquid release values in Table R-1 represent two times the appropriate ODCM release rate limits associated with the specified monitors (ref. 1, 2).

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

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

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

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

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

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

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

CNP Basis Reference(s):

1. EP-CALC-CNP-1601, Radiological Effluent EAL Threshold Values
2. PMP-6010-OSD-001, Off Site Dose Calculation Manual
3. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

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

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

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

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

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

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

CNP Basis Reference(s):

1. PMP-6010-OSD-001, Off Site Dose Calculation Manual
2. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.1 Alert

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

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

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

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

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

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	ALERT	UE
Gaseous	Unit Vent Noble Gas	VRS-1500 (2500)	3.3E+00 µCi/cc	3.3E-01 µCi/cc	3.3E-02 µCi/cc	4.2E-03 µCi/cc
Gaseous	Gland Seal Leakoff	SRA-1800 (2800)	1.6E+02 µCi/cc	1.6E+01 µCi/cc	1.6E+00 µCi/cc	1.4E-01 µCi/cc
Gaseous	Steam Jet Air Ejector	SRA-1900 (2900)	1.5E+04 µCi/cc	1.5E+03 µCi/cc	1.5E+02 µCi/cc	1.3E+01 µCi/cc
Liquid	Radwaste Effluent	RRS-1001	—	—	—	4.6E+04 cpm
Liquid	SG Blowdown	R-19	—	—	—	1.7E+03 cpm
Liquid	SG Blowdown	DRS-3100/4100	—	—	—	1.2E+04 cpm
Liquid	SG Blowdown Treatment	R-24	—	—	—	2.9E+04 cpm
Liquid	SG Blowdown Treatment	DRS-3200/4200	—	—	—	1.2E+05 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either:

- 10 mrem TEDE
- 50 mrem CDE Thyroid

The column "ALERT" gaseous effluent release values in Table R-1 correspond to calculated doses of 1% (10% of the SAE thresholds) of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

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

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

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

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

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

CNP Basis Reference(s):

1. EP-CALC-CNP-1601, Radiological Effluent EAL Threshold Values
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

Dose assessments are performed by computer-based or manual methods (ref. 1).

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

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

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

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

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

CNP Basis Reference(s):

1. PMP-2080-EPP-108 Initial Dose Assessment
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.3	**Alert**
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the site boundary for 60 min. of exposure (Notes 1, 2)	

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

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

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

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

CNP Basis Reference(s):

1. PMP-6010-OSD-001, Off Site Dose Calculation Manual
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.4 Alert

Field survey results indicate **EITHER** of the following at or beyond the site boundary:

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

RMT-2080-EOF-001, Activation and Operation of the EOF provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

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

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

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

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

CNP Basis Reference(s):

1. RMT-2080-EOF-001 Activation and Operation of the EOF
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.1 Site Area Emergency

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

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

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

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

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

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	ALERT	UE
Gaseous	Unit Vent Noble Gas	VRS-1500 (2500)	3.3E+00 μCi/cc	3.3E-01 μCi/cc	3.3E-02 μCi/cc	4.2E-03 μCi/cc
Gaseous	Gland Seal Leakoff	SRA-1800 (2800)	1.6E+02 μCi/cc	1.6E+01 μCi/cc	1.6E+00 μCi/cc	1.4E-01 μCi/cc
Gaseous	Steam Jet Air Ejector	SRA-1900 (2900)	1.5E+04 μCi/cc	1.5E+03 μCi/cc	1.5E+02 μCi/cc	1.3E+01 μCi/cc
Liquid	Radwaste Effluent	RRS-1001	—	—	—	4.6E+04 cpm
Liquid	SG Blowdown	R-19	—	—	—	1.7E+03 cpm
Liquid	SG Blowdown	DRS-3100/4100	—	—	—	1.2E+04 cpm
Liquid	SG Blowdown Treatment	R-24	—	—	—	2.9E+04 cpm
Liquid	SG Blowdown Treatment	DRS-3200/4200	—	—	—	1.2E+05 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either:

- 100 mrem TEDE
- 500 mrem CDE Thyroid

The column "SAE" gaseous effluent release value in Table R-1 corresponds to calculated doses of 10% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

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

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

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

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

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

CNP Basis Reference(s):

1. EP-CALC-CNP-1601-CNP-1601, Radiological Effluent EAL Threshold Values
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.2	**Site Area Emergency**
Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the site boundary (Note 4)	

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

Mode Applicability:

All

Definition(s):

None

Basis:

Dose assessments are performed by computer-based and manual methods (ref. 1).

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

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

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

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

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

CNP Basis Reference(s):

1. PMP-2080-EPP-108 Initial Dose Assessment
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.3 Site Area Emergency

Field survey results indicate **EITHER** of the following at or beyond the site boundary:

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

RMT-2080-EOF-001, Activation and Operation of the EOF provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

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

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

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

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

CNP Basis Reference(s):

1. RMT-2080-EOF-001 Activation and Operation of the EOF
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.1 General Emergency

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

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

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

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

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

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	ALERT	UE
Gaseous	Unit Vent Noble Gas	VRS-1500 (2500)	3.3E+00 µCi/cc	3.3E-01 µCi/cc	3.3E-02 µCi/cc	4.2E-03 µCi/cc
Gaseous	Gland Seal Leakoff	SRA-1800 (2800)	1.6E+02 µCi/cc	1.6E+01 µCi/cc	1.6E+00 µCi/cc	1.4E-01 µCi/cc
Gaseous	Steam Jet Air Ejector	SRA-1900 (2900)	1.5E+04 µCi/cc	1.5E+03 µCi/cc	1.5E+02 µCi/cc	1.3E+01 µCi/cc
Liquid	Radwaste Effluent	RRS-1001	—	—	—	4.6E+04 cpm
Liquid	SG Blowdown	R-19	—	—	—	1.7E+03 cpm
		DRS-3100/4100	—	—	—	1.2E+04 cpm
	SG Blowdown Treatment	R-24	—	—	—	2.9E+04 cpm
		DRS-3200/4200	—	—	—	1.2E+05 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either:

- 1000 mrem TEDE
- 5000 mrem CDE Thyroid

The column "GE" gaseous effluent release values in Table R-1 correspond to calculated doses of 100% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

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

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

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

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

CNP Basis Reference(s):

1. EP-CALC-CNP-1601, Radiological Effluent EAL Threshold Values
2. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

Dose assessments are performed by computer-based and manual methods (ref. 1).

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

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

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

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

CNP Basis Reference(s):

1. PMP-2080-EPP-108 Initial Dose Assessment
2. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.3 General Emergency

Field survey results indicate **EITHER** of the following at or beyond the site boundary:

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

RMT-2080-EOF-001, Activation and Operation of the EOF provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

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

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

CNP Basis Reference(s):

1. RMT-2080-EOF-001 Activation and Operation of the EOF
2. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Unplanned loss of water level above irradiated fuel

EAL:

RU2.1 Unusual Event

UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication

AND

UNPLANNED rise in corresponding area radiation levels as indicated on **any** of the following radiation monitors:

- VRS-1101/1201, Unit 1 Upper Containment
- VRS-2101/2201, Unit 2 Upper Containment
- R-5 Spent Fuel Area
- VRS-5006 Spent Fuel Area

Mode Applicability:

All

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

REFUELING PATHWAY-. The reactor refueling cavity, spent fuel pool and fuel transfer canal comprise the refueling pathway.

Basis:

The low water level alarm in this EAL refers to the Spent Fuel Pool (SFP) low (Panel 134 Drop 2) or low-low level alarms (Panel 105 Drop 27 or Panel 205 Drop 27) (ref. 1). During the fuel transfer phase of refueling operations, the fuel transfer canal is normally in communication with the spent fuel pool and the refueling cavity in the Containment is in communication with the fuel transfer canal when the fuel transfer tube is open. A lowering in water level in the SFP, fuel transfer canal or refueling cavity is therefore sensed by the SFP low level alarm. Neither the refueling cavity nor the fuel transfer canal is equipped with a low level alarm (ref. 1).

Technical Specification Section 3.7.14 (ref. 5) requires at least 23 ft of water above the SFP storage racks. Technical Specification Section 3.9.6 (ref. 4) requires at least 23 ft of water above the Reactor Vessel flange in the refueling cavity. During refueling, this maintains sufficient water level in the fuel transfer canal, refueling cavity, and SFP to retain iodine fission product activity in the water in the event of a fuel handling accident.

The listed radiation monitors are those expected to see increase area radiation levels as a result of a loss of REFUELING PATHWAY inventory (ref. 1, 2, 3). Increasing radiation indications on these monitors in the absence of indications of decreasing REFUELING PATHWAY level are not classifiable under this EAL.

When the spent fuel pool and reactor cavity are connected, there could exist the possibility of uncovering irradiated fuel. Therefore, this EAL is applicable for conditions in which irradiated fuel is being transferred to and from the reactor vessel and spent fuel pool.

This IC addresses a decrease in water level above irradiated fuel sufficient to cause elevated radiation levels. This condition could be a precursor to a more serious event and is also indicative of a minor loss in the ability to control radiation levels within the plant. It is therefore a potential degradation in the level of safety of the plant.

A water level decrease will be primarily determined by indications from available level instrumentation. Other sources of level indications may include reports from plant personnel (e.g., from a refueling crew) or video camera observations (if available). A significant drop in the water level may also cause an increase in the radiation levels of adjacent areas that can be detected by monitors in those locations.

The effects of planned evolutions should be considered. For example, a refueling bridge area radiation monitor reading may increase due to planned evolutions such as lifting of the reactor vessel head or movement of a fuel assembly. Note that this EAL is applicable only in cases where the elevated reading is due to an unplanned loss of water level.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC RA2.

CNP Basis Reference(s):

1. 12-OHP-0422-018-002 Loss of Refueling Water Level During Refueling Operations – Local Actions
2. 12-OHP-4022-018-003 Irradiated Fuel Handling Accident in Containment Building – Local Actions
3. 12-OHP-4022-018-004 Irradiated Fuel Handling Accident in Containment Building – Control Room Actions
4. Technical Specification Section 3.9.6 Refueling Cavity Water Level
5. Technical Specification Section 3.7.14 Fuel Storage Pool Water Level
6. NEI 99-01 AU2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel

EAL:

RA2.1	**Alert**
Uncovery of irradiated fuel in the REFUELING PATHWAY	

Mode Applicability:

All

Definition(s):

REFUELING PATHWAY - The reactor refueling cavity, spent fuel pool and fuel transfer canal comprise the refueling pathway.

Basis:

This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant. This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with EU1.1.

Escalation of the emergency would be based on either Recognition Category R or C ICs.

This EAL escalates from RU2.1 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used (e.g., a boil-off curve). Classification of an event using this EAL should be based on the totality of available indications, reports and observations.

While an area radiation monitor could detect an increase in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance with Recognition Category C during the Cold Shutdown and Refueling modes.

CNP Basis Reference(s):

1. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel

EAL:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by High alarm on **any** of the following radiation monitors:

- VRS-1101/1201, Unit 1 Upper Containment
- VRS-2101/2201, Unit 2 Upper Containment
- R-5 Spent Fuel Area
- VRS-5006 Spent Fuel Area

Mode Applicability:

All

Definition(s):

None

Basis:

The specified radiation monitors are those expected to see increase area radiation levels as a result of damage to irradiated fuel (ref. 1, 2, 3, 4).

This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with EU1.1.

Escalation of the emergency would be based on either Recognition Category R or C ICs.

This EAL addresses a release of radioactive material caused by mechanical damage to irradiated fuel. Damaging events may include the dropping, bumping or binding of an assembly, or dropping a heavy load onto an assembly. A rise in readings on radiation monitors should be considered in conjunction with in-plant reports or observations of a potential fuel damaging event (e.g., a fuel handling accident).

CNP Basis Reference(s):

1. 12-OHP-4022-018-003 Irradiated Fuel Handling Accident in Containment Building – Local Actions

2. 12-OHP-4022-018-004 Irradiated Fuel Handling Accident in Containment Building – Control Room Actions
3. 12-OHP-4022-018-005 Irradiated Fuel Handling Accident in Spent Fuel Storage Area – Local Actions
4. 12-OHP-4022-018-006 Irradiated Fuel Handling Accident in Spent Fuel Storage Area – Control Room Actions
5. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel

EAL:

RA2.3 Alert

Lowering of spent fuel pool level to 9 ft. 6 in. on 1(2)-RLI-502-CRI Spent Fuel Pit Level Indication (8 ft. 10 in. on local ruler)

Mode Applicability:

All

Definition(s):

None

Basis:

Post-Fukushima order EA-12-051 (ref. 1) required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3) (ref. 1).

For CNP SFP Level 2 is plant elevation 630 ft. 10.5 in. or 9 ft. 6 in. as indicated on 1(2)-RLI-502-CRI in the Control Room or 1(2)-RLI-502-BATT back-up indicator (ref. 2). This level corresponds to 8 ft. 10 in. on the SFP ruler.

This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

Spent fuel pool water level at this value is within the lower end of the level range necessary to prevent significant dose consequences from direct gamma radiation to personnel performing operations in the vicinity of the spent fuel pool. This condition reflects a significant loss of spent fuel pool water inventory and thus it is also a precursor to a loss of the ability to adequately cool the irradiated fuel assembles stored in the pool.

Escalation of the emergency classification level would be via IC RS1or RS2.

CNP Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. EC-0000052892 Spent Fuel Pool Level for NRC Order EA-12-051
3 NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Spent fuel pool level at the top of the fuel racks

EAL:

RS2.1	**Site Area Emergency**
Lowering of spent fuel pool level to 0 ft. on 1(2)-RLI-502-CRI Spent Fuel Pit Level Indication	

Mode Applicability:

All

Definition(s):

None

Basis:

Post-Fukushima order EA-12-051 (ref. 1) required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3) (ref. 1).

For CNP SFP Level 3 is plant elevation 620 ft. 10.5 in. However, the SFP level instrument lower range (0 ft.) corresponds to plant elevation 621 ft. 6 in. Therefore an indicated level of 0 ft. on 1(2)-RLI-502-CRI in the Control Room or 1(2)-RLI-502-BATT back-up indicator is used as indicated Level 3 (ref. 2).

This EAL addresses a significant loss of spent fuel pool inventory control and makeup capability leading to IMMINENT fuel damage. This condition entails major failures of plant functions needed for protection of the public and thus warrant a SITE AREA EMERGENCY declaration.

It is recognized that this IC would likely not be met until well after another SITE AREA EMERGENCY IC was met; however, it is included to provide classification diversity.

Escalation of the emergency classification level would be via IC RG1 or RG2.

CNP Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. EC-0000052892 Spent Fuel Pool Level for NRC Order EA-12-051
3. NEI 99-01 AS2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer

EAL:

RG2.1	**General Emergency**
Spent fuel pool level **cannot** be restored to at least 0 ft. on 1(2)-RLI-502-CRI Spent Fuel Pit Level Indication for ≥ 60 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

All

Definition(s):

None

Basis:

Post-Fukushima order EA-12-051 (ref. 1) required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3) (ref. 1).

For CNP SFP Level 3 is plant elevation 620 ft. 10.5 in. However, the SFP level instrument lower range (0 ft.) corresponds to plant elevation 621 ft. 6 in. Therefore an indicated level of 0 ft. on 1(2)-RLI-502-CRI in the Control Room or 1(2)-RLI-502-BATT back-up indicator is used as indicated Level 3 (ref. 2).

This EAL addresses a significant loss of spent fuel pool inventory control and makeup capability leading to a prolonged uncovery of spent fuel. This condition will lead to fuel damage and a radiological release to the environment.

It is recognized that this IC would likely not be met until well after another GENERAL EMERGENCY IC was met; however, it is included to provide classification diversity.

CNP Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. EC-0000052892 Spent Fuel Pool Level for NRC Order EA-12-051
3. NEI 99-01 AG2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

Initiating Condition: Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

RA3.1 Alert

Dose rates > 15 mR/hr in **any** of the following areas:

- Unit 1 Control Room (ERS-7401)
- Unit 2 Control Room (ERS-8401)
- Central Alarm Station (by survey)
- Secondary Alarm Station (by survey)

Mode Applicability:

All

Definition(s):

None

Basis:

Areas that meet this threshold include the Control Rooms and the Central Alarm Station (CAS) and Secondary Alarm Station (SAS). ERS-7401 (ERS-8401) monitor the Control Rooms for area radiation (ref. 1). The CAS and SAS are included in this EAL because of their importance to permitting access to areas required to assure safe plant operations (ref. 1).

There is no permanently installed CAS or SAS area radiation monitors that may be used to assess this EAL threshold. Therefore this threshold must be assessed via local radiation survey for the CAS and SAS (ref. 1).

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The SEC should consider the cause of the increased radiation levels and determine if another IC may be applicable.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

CNP Basis Reference(s):

1. FSAR Table 11.3-1 Radiation Monitoring System Channel Sensitivities and Detecting Medium
2. NEI 99-01 AA3

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

Initiating Condition: Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

RA3.2 Alert

An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 rooms or areas (Note 5)

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

Table R-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Auxiliary Building 573'	4, 5
Auxiliary Building 587' (including D/G room)	1, 2, 3, 4, 5
Auxiliary Building 591'	1, 2, 3, 4
Auxiliary Building 609' (including 4kV room)	1, 2, 3, 4, 5
Auxiliary Building 633'	1, 2, 3, 4
Turbine Building (**All** Levels)	1, 2, 3
Turbine Building 591'	4, 5
Screenhouse	1, 2, 3, 4, 5

Mode Applicability:

All

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

The list of plant rooms or areas with entry-related mode applicability identified specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The SEC should consider the cause of the increased radiation levels and determine if another IC may be applicable. For this EAL, an ALERT declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the increased radiation levels. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., installing temporary shielding, requiring use of non-routine protective equipment, requesting an extension in dose limits beyond normal administrative limits).

An emergency declaration is not warranted if any of the following conditions apply.

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the elevated radiation levels). For example, the plant is in Mode 1 when the radiation increase occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The increased radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

CNP Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Rooms/Areas Tables R-2 & H-2 Bases
2. NEI 99-01 AA3

Category E – Independent Spent Fuel Storage Installation (ISFSI)

EAL Group: Any (EALs in this category are applicable to any plant condition, hot or cold.)

An independent spent fuel storage installation (ISFSI) is a complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage. A significant amount of the radioactive material contained within a canister must escape its packaging and enter the biosphere for there to be a significant environmental effect resulting from an accident involving the dry storage of spent nuclear fuel.

An UNUSUAL EVENT is declared on the basis of the occurrence of an event of sufficient magnitude that a loaded cask confinement boundary is damaged or violated.

Category: ISFSI

Subcategory: Confinement Boundary

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Unusual Event

Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading:

- > 60 mrem/hr (gamma + neutron) on the top of the overpack
- > 600 mrem/hr gamma + neutron) on the side of the overpack excluding inlet and outlet ducts

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY-. The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As applied to the CNP ISFSI, the CONFINEMENT BOUNDARY is defined to be the Multi-Purpose Canister (MPC).

Basis:

Overpacks are the HI-STORM 100 casks which receive and contain the sealed MPCs for interim storage in the ISFSI. They provide gamma and neutron shielding, and provide for ventilated air flow to promote heat transfer from the MPC to the environs. The term overpack does not include the transfer cask (ref. 1).

The value shown represents 2 times the maximum overpack surface dose rates specified in Section 5.7 of the ISFSI Certificate of Compliance Technical Specifications for radiation external to a loaded MPC overpack (ref. 1).

This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage.

The existence of "damage" is determined by radiological survey. The technical specification multiple of "2 times", which is also used in Recognition Category R IC RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

Security-related events for ISFSIs are covered under ICs HU1 and HA1.

CNP Basis Reference(s):

1. Certificate of Compliance No. 1014 Holtec International HI-STORM 100 Cask System Safety Evaluation Report Amendment 1 Appendix A Technical Specifications Section 5.7
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 - Refueling, D – Defueled).

The events of this category pertain to the following subcategories:

1. RCS Level

RCS water level is directly related to the status of adequate core cooling and, therefore, fuel clad integrity.

2. Loss of Emergency AC Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite power sources for 4.16KV AC emergency buses.

3. RCS Temperature

Uncontrolled or inadvertent temperature or pressure increases are indicative of a potential loss of safety functions.

4. Loss of Vital DC Power

Loss of emergency 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 vital plant 250 VDC power sources.

5. Loss of Communications

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

6. Hazardous Event Affecting Safety Systems

Certain hazardous natural and technological events may result in VISIBLE DAMAGE to or degraded performance of safety systems warranting classification.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory

EAL:

CU1.1	**Unusual Event**
UNPLANNED loss of reactor coolant results in RCS water level less than a required lower limit for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

With the plant in Cold Shutdown, RCS water level is normally established by 1 (2)-OHP-4021-002-005, RCS Draining (ref. 1). If RCS level is being controlled below the pressurizer low level setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

With the plant in Refueling mode, RCS water level is normally maintained at or above the reactor vessel flange (Technical Specification LCO 3.9.6 requires at least 23 ft. of water above the top of the reactor vessel flange in the refueling cavity during refueling operations) (ref. 2).

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor RCS level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an UNUSUAL EVENT due to the reduced water inventory that is available to keep the core covered.

This EAL recognizes that the minimum required RCS level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.

The 15-minute threshold duration allows sufficient time for prompt operator actions to restore and maintain the expected water level. This criterion excludes transient conditions causing a brief lowering of water level.

Continued loss of RCS inventory may result in escalation to the ALERT emergency classification level via either IC CA1 or CA3.

CNP Basis Reference(s):

1. 1(2)-OHP-4021-002-005, RCS Draining
2. Technical Specification Section 3.9.6 Refueling Cavity Water Level
3. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory

EAL:

CU1.2 Unusual Event

RCS water level cannot be monitored

AND EITHER

- UNPLANNED increase in **any** Table C-1 sump/tank level due to loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

Table C-1 Sumps / Tanks
• Containment Sumps • Auxiliary Building Sumps • RWST • RCDT

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available.

In this EAL, all water level indication is unavailable and the RCS inventory loss must be detected by indirect leakage indications. 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. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor RCS level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an UNUSUAL EVENT due to the reduced water inventory that is available to keep the core covered.

This EAL addresses a condition where all means to determine level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels (Table C-1). Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

Continued loss of RCS inventory may result in escalation to the ALERT emergency classification level via either IC CA1 or CA3.

CNP Basis Reference(s):

1. 1(2)-OHP-4022-002-020 Excessive Reactor Coolant Leakage
2. 1(2)-OHP-4021-002-005, RCS Draining
3. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Significant loss of RCS inventory

EAL:

CA1.1	**Alert**
Loss of RCS inventory as indicated by RCS level < 614.0 ft.	

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

None

Basis:

614.0 ft. corresponds to midloop and is the minimum allowed RCS level for operation of RHR (ref.1)

RCS level cannot be measured below 612 feet on NLI-1000, High Resolution – RCS Full Range Level Indication, which is below the bottom ID of the hot leg inlet. Should RCS level drop below this point it is assumed water level cannot be monitored other than visually.

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

For this EAL, a lowering of RCS water level below 614.0 ft. indicates that operator actions have not been successful in restoring and maintaining RCS water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

Although related, this EAL is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Residual Heat Removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

If RCS water level continues to lower, then escalation to SITE AREA EMERGENCY would be via IC CS1.

CNP Basis Reference(s):

1. 1(2)-OHP-4022-017-001 Loss of RHR Cooling
2. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Significant loss of RCS inventory

EAL:

CA1.2 Alert

RCS water level cannot be monitored for ≥ 15 min. (Note 1)

AND EITHER

- UNPLANNED increase in **any** Table C-1 sump/tank level due to loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-1 Sumps / Tanks
• Containment Sumps
• Auxiliary Building Sumps
• RWST
• RCDT

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available.

In the Refuel mode, the RCS is not intact and RPV level may be monitored by different means, including the ability to monitor level visually.

In this EAL, all RCS water level indication would be unavailable for greater than 15 minutes, and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). Surveillance procedures provide instructions for calculating primary system leak rate by manual or computer-based water inventory balances. 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. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

For this EAL, the inability to monitor RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

The 15-minute duration for the loss of level indication was chosen because it is half of the EAL duration specified in IC CS1.

If the RCS inventory level continues to lower, then escalation to SITE AREA EMERGENCY would be via IC CS1.

CNP Basis Reference(s):

1. 1(2)-OHP-4022-002-020 Excessive Reactor Coolant Leakage
2. 1(2)-OHP-4021-002-005, RCS Draining
3. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory affecting core decay heat removal capability

EAL:

CS1.1 Site Area Emergency

RCS water level cannot be monitored for ≥ 30 min. (Note 1)

AND

Core uncovery is indicated by **any** of the following:

- UNPLANNED increase in **any** Table C-1 sump/tank level of sufficient magnitude to indicate core uncovery
- High alarm on Containment radiation monitor VRA-1310 (2310) or VRA-1410(2410)
- Erratic Source Range Monitor indication

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-1 Sumps / Tanks
• Containment Sumps • Auxiliary Building Sumps • RWST • RCDT

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available.

In the Refueling mode, the RCS is not intact and reactor vessel level may be monitored by different means, including the ability to monitor level visually.

In this EAL, all RCS water level indication would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). Surveillance procedures provide instructions for calculating primary system leak rate by manual or computer-based water inventory balances. 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. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

The reactor vessel inventory loss may be detected by the containment radiation monitors VRA-1310 (2310) or 1410 (2410) or erratic Source Range Monitor indication. 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 a high alarm on containment high range radiation monitors (ref. 3).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations (ref. 4).

This IC addresses a significant and prolonged loss of RCS inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a SITE AREA EMERGENCY declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RCS level cannot be restored, fuel damage is probable.

The 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

This EAL addresses concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal; SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues; NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States; and NUMARC-91-06, Guidelines for Industry Actions to Assess Shutdown Management.

Escalation of the emergency classification level would be via IC CG1 or RG1

CNP Basis Reference(s):

1. 1(2)-OHP-4022-002-020 Excessive Reactor Coolant Leakage
2. 1(2)-OHP-4021-002-005, RCS Draining
3. Calculation No. 1-2-UNC-421 Post Accident High Range Containment Area Radiation Monitoring Loop Uncertainty Calculation
4. Nuclear Safety Analysis Center (NSAC), 1980, “Analysis of Three Mile Island - Unit 2 Accident,” NSAC-1
5. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory affecting fuel clad integrity with containment challenged

EAL:

CG1.1 General Emergency

RCS level **cannot** be monitored for ≥ 30 min. (Note 1)

AND

Core uncovery is indicated by **any** of the following:

- UNPLANNED increase in **any** Table C-1 sump/tank level of sufficient magnitude to indicate core uncovery
- High alarm on Containment radiation monitor VRA-1310 (2310) or VRA-1410(2410)
- Erratic Source Range Monitor indication

AND

Any Containment Challenge indication, Table C-2

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a GENERAL EMERGENCY is not required.

Table C-1 Sumps / Tanks
• Containment Sumps • Auxiliary Building Sumps • RWST • RCDT

Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • Containment hydrogen concentration ≥ 4% • Unplanned rise in Containment pressure

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to CNP, Containment Closure is established when the requirements of PMP-4100-SDR-001 are met.

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available.

In the Refueling mode, the RCS is not intact and RPV level may be monitored by different means, including the ability to monitor level visually.

In this EAL, all RCS water level indication would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). Surveillance procedures provide instructions for calculating primary system leak rate by manual or computer-based water inventory balances. 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. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

The reactor vessel inventory loss may be detected by the containment radiation monitors VRA-1310 (2310) or 1410 (2410) or erratic Source Range Monitor indication. 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 a high alarm on containment high range radiation monitors (ref. 3).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations (ref. 4).

Three conditions are associated with a challenge to containment integrity:

1. CONTAINMENT CLOSURE not established - The status of Containment closure is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 5). If containment closure is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to GE would not occur.
2. Containment hydrogen ≥ 4% - The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations. CNP is equipped with a Post-Accident Containment Hydrogen Monitoring System (PACHMS) that is capable of

continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident for accident mitigation, including emergency planning. PACHMS is comprised of two sampling-analyzing-control trains. Each train has two subsystems - the hydrogen analyzer panels and the remote control panels (ref. 6).

3. UNPLANNED rise in Containment pressure - An unplanned pressure rise in containment while in cold Shutdown or Refueling modes can threaten Containment Closure capability and thus containment potentially cannot be relied upon as a barrier to fission product release.

This IC addresses the inability to restore and maintain reactor vessel level above the top of active fuel with containment challenged. This condition represents actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RCS level cannot be restored, fuel damage is probable.

With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a GENERAL EMERGENCY is not required.

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to Containment integrity.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.

The 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

This EAL addresses concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.*

CNP Basis Reference(s):

1. 1(2)-OHP-4022-002-020 Excessive Reactor Coolant Leakage
2. 1(2)-OHP-4021-002-005, RCS Draining
3. Calculation No. 1-2-UNC-421 Post Accident High Range Containment Area Radiation Monitoring Loop Uncertainty Calculation
4. Nuclear Safety Analysis Center (NSAC), 1980, "Analysis of Three Mile Island - Unit 2 Accident," NSAC-1
5. PMP-4100-SDR-001 Plant Shutdown Safety and Risk Management
6. UFSAR Section 7.8.2 Post-Accident Containment Hydrogen Monitoring
7. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

Initiating Condition: Loss of all but one AC power source to emergency buses for 15 minutes or longer

EAL:

CU2.1 Unusual Event

AC power capability, Table C-3, to emergency 4.16 kV buses T11A (T21A) and T11D (T21D) reduced to a single power source for ≥ 15 min. (Note 1)

AND

Any additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-3 AC Power Sources
Offsite: • Reserve Auxiliary Xmr TR101AB (TR201AB) • Reserve Auxiliary Xmr TR101CD (TR201CD) • 69/4.16 kV Alternate Xmr TR12EP-1 • Main Xmr TR1 (TR2) backfeed (only if already aligned) **Onsite:** • EDG 1AB (2AB) • EDG 1CD (2CD)

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling, D - Defueled

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the emergency buses, whether or not the buses are powered from it.

The condition indicated by this EAL is the degradation of the offsite and onsite power sources such that any additional single failure would result in a loss of all AC power to the emergency buses.

A list of onsite and offsite AC power sources credited for this EAL are specified in Table C-3.

4.16KV buses T11A (T21A) and T11D (T21D) are the emergency (essential) buses (ref. 1). While generating, auxiliary power is normally supplied from the generator terminals through the unit auxiliary transformers (TR1AB and TR1CD for Unit 1 and TR2AB and TR2CD for Unit 2). When the plant trips or the plant is shutdown the station auxiliaries are transferred to the preferred offsite power source (that is, to reserve auxiliary transformers TR101AB and TR101CD for Unit 1 and TR201AB and TR201CD for Unit 2) to assure continued power to equipment when the main generator is off-line (ref. 1).

In addition, an alternate offsite power source, a 69/4.16kV transformer (TR12EP-1), located at the plant site, has the necessary capacity to operate one train of the engineered safeguard equipment in one unit while supplying one train of the safe shutdown power in the other.

T11A (T21A) and T11D (T21D) also each have an emergency diesel generator which supply onsite electrical power to the bus automatically in the event that the preferred offsite sources become unavailable (ref. 1).

Another method to obtain offsite power is by backfeeding the emergency buses through the main transformer and unit auxiliary transformers. This is only done during cold shutdown when no other power sources are available (ref. 1, 3). Credit is only taken for this source if already aligned as it requires removal of the main generator disconnect links.

The Supplemental Diesel Generators (SDGs) are not credited as an AC power source for this EAL.

This cold condition EAL is equivalent to the hot condition EAL SA1.1.

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as an ALERT because of the increased time available to restore another power source to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an essential bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel

generators) with a single train of emergency buses being back-fed from the unit main generator.

- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from an offsite power source.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

The subsequent loss of the remaining single power source would escalate the event to an ALERT in accordance with IC CA2.

CNP Basis Reference(s):

1. UFSAR Figure 8.1-1A(B) Main Auxiliary One-Line Diagram
2. UFSAR Section 8.0 Electrical Systems
3. 1(2)-OHP-4022-001-005 Loss of Offsite Power with Reactor Shutdown
4. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite and **all** onsite AC power to emergency buses for 15 minutes or longer

EAL:

CA2.1	**Alert**
Loss of **all** offsite and **all** onsite AC power to emergency 4.16KV buses T11A (T21A) and T11D (T21D) for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, D - Defueled

Basis:

4.16KV buses T11A (T21A) and T11D (T21D) are the emergency (essential) buses (ref. 1). While generating, auxiliary power is normally supplied from the generator terminals through the unit auxiliary transformers (TR1AB and TR1CD for Unit 1 and TR2AB and TR2CD for Unit 2). When the plant trips or the plant is shutdown the station auxiliaries are transferred to the preferred offsite power source (that is, to reserve auxiliary transformers TR101AB and TR101CD for Unit 1 and TR201AB and TR201CD for Unit 2) to assure continued power to equipment when the main generator is off-line (ref. 1).

In addition, an alternate offsite power source, a 69/4.16kV transformer (TR12EP-1), located at the plant site, has the necessary capacity to operate one train of the engineered safeguard equipment in one unit while supplying one train of the safe shutdown power in the other.

T11A (T21A) and T11D (T21D) also each have an emergency diesel generator which supply electrical power to the bus automatically in the event that the preferred offsite sources become unavailable (ref. 1).

Another method to obtain offsite power is by backfeeding the emergency buses through the main transformer and unit auxiliary transformers. This is only done during cold shutdown when no other power sources are available (ref. 1, 3). Credit is only taken for this source if already aligned as it requires removal of the main generator disconnect links.

The Supplemental Diesel Generators (SDGs) or any other alternative AC power source capable of powering an emergency bus can also be credited as an AC power source for this EAL.

This cold condition EAL is equivalent to the hot condition loss of all offsite AC power EAL SS1.1.

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as a SITE AREA EMERGENCY because of the increased time available to restore an emergency bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition represents an actual or potential substantial degradation of the level of safety of the plant.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via IC CS1 or RS1.

CNP Basis Reference(s):

1. UFSAR Figure 8.1-1A(B) Main Auxiliary One-Line Diagram
2. UFSAR Section 8.0 Electrical Systems
3. 1(2)-OHP-4022-001-005 Loss of Offsite Power with Reactor Shutdown
4. NEI 99-01 CA2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.1	**Unusual Event**
UNPLANNED increase in RCS temperature to > 200°F (Note 10)	

Note 10: Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss of decay heat removal.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

The following instrumentation is capable of providing indication of an RCS temperature rise that approaches the Technical Specification cold shutdown temperature limit of (200° F) (ref. 1, 2):

- NTI-100, NTI-101, Selected Incore Temperature or Temporary Thermocouples
- NTR-210, Reactor Coolant T-Cold Wide Range Loop 1
- NTR-220, Reactor Coolant T-Cold Wide Range Loop 2
- NTR-230, Reactor Coolant T-Cold Wide Range Loop 3
- NTR-240, Reactor Coolant T-Cold Wide Range Loop 4
- NTR-110, Reactor Coolant T-Hot Loop 1
- NTR-120, Reactor Coolant T-Hot Loop 2
- NTR-130, Reactor Coolant T-Hot Loop 3
- NTR-140, Reactor Coolant T-Hot Loop 4
- RHR display on PPC

In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on time to boil data (ref.2).

This IC addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit and represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the SEC should also refer to IC CA3.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

This EAL involves a loss of decay heat removal capability, or an addition of heat to the RCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

During an outage, the level in the reactor vessel will normally be maintained at or above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid increase in reactor coolant temperature depending on the time after shutdown.

Escalation to ALERT would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

CNP Basis Reference(s):

1. 1(2)-OHP-4021-001-004, Plant Cooldown from Hot Standby to Cold Shutdown
2. 1(2)-OHP-4022-017-001, Loss of RHR Cooling
3. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.2 Unusual Event

Loss of **all** RCS temperature and RCS level indication for ≥ 15 min. (Note 1)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6- Refueling

Definition(s):

None

Basis:

The following instrumentation is capable of providing indication of an RCS temperature rise that approaches the Technical Specification cold shutdown temperature limit of (200° F) (ref. 1, 2):

- NTI-100, NTI-101, Selected Incore Temperature or Temporary Thermocouples
- NTR-210, Reactor Coolant T-Cold Wide Range Loop 1
- NTR-220, Reactor Coolant T-Cold Wide Range Loop 2
- NTR-230, Reactor Coolant T-Cold Wide Range Loop 3
- NTR-240, Reactor Coolant T-Cold Wide Range Loop 4
- NTR-110, Reactor Coolant T-Hot Loop 1
- NTR-120, Reactor Coolant T-Hot Loop 2
- NTR-130, Reactor Coolant T-Hot Loop 3
- NTR-140, Reactor Coolant T-Hot Loop 4
- RHR display on PPC

RCS level indications include pressurizer level, narrow and wide range RVLIS and RC Loop narrow, mid and wide range instruments, NGG-100 and Mansell level instrument (ref. 2.3.4).

This EAL addresses the inability to determine RCS temperature and level, and represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the SEC should also refer to IC CA3.

This EAL reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation to ALERT would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

CNP Basis Reference(s):

1. 1(2)-OHP-4021-001-004, Plant Cooldown from Hot Standby to Cold Shutdown
2. 1(2)-OHP-4022-017-001, Loss of RHR Cooling
3. 1(2)-OHP-4022-002-020 Excessive Reactor Coolant Leakage
4. 1(2)-OHP-4021-002-005, RCS Draining
5. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.1 Alert

UNPLANNED increase in RCS temperature to > 200°F for > Table C-4 duration (Notes 1, 10)

OR

UNPLANNED RCS pressure increase > 10 psig (This EAL does not apply during water-solid plant conditions

Note 1: The SEC should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

Note 10: Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss of decay heat removal.

Table C-4: RCS Heat-up Duration Thresholds

RCS Status	CONTAINMENT CLOSURE Status	Heat-up Duration
INTACT (but **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 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to CNP, Containment Closure is established when the requirements of PMP-4100-SDR-001 are met.

UNPLANNED -. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

INTACT (RCS) - The RCS should be considered intact when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams).

REDUCED INVENTORY - Operating condition when fuel is in the reactor vessel and Reactor Coolant System level is lower than 3 feet (or more) below the Reactor Vessel flange.

Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1). These include (ref. 2, 3):

- NTI-100, NTI-101, Selected Incore Temperature or Temporary Thermocouples
- NTR-210, Reactor Coolant T-Cold Wide Range Loop 1
- NTR-220, Reactor Coolant T-Cold Wide Range Loop 2
- NTR-230, Reactor Coolant T-Cold Wide Range Loop 3
- NTR-240, Reactor Coolant T-Cold Wide Range Loop 4
- NTR-110, Reactor Coolant T-Hot Loop 1
- NTR-120, Reactor Coolant T-Hot Loop 2
- NTR-130, Reactor Coolant T-Hot Loop 3
- NTR-140, Reactor Coolant T-Hot Loop 4
- RHR display on PPC

The following instrumentation is capable of providing indication of a 10 psig rise in RCS pressure:

- NLI-1000A/B, RCS Pressure
- NLI-122A/B (MLMS Cart C), RCS Pressure
- NPS-110 (Loop 1) Reactor Vessel Train A Wide Range Pressure
- NPS-111 (Loop 3) Reactor Vessel Train B Wide Range Pressure

In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the RCS pressure increase criteria when in Mode 5 or based on time to boil data when in Mode 6 (ref. 3).

This IC addresses conditions involving a loss of decay heat removal capability or an addition of heat to the RCS in excess of that which can currently be removed. Either condition represents an actual or potential substantial degradation of the level of safety of the plant.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

The RCS Heat-up Duration Thresholds table addresses an increase in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not INTACT, or RCS inventory is reduced (e.g., mid-loop operation). The 20-minute criterion was included to allow time for operator action to address the temperature increase.

The RCS Heat-up Duration Thresholds table also addresses an increase in RCS temperature with the RCS INTACT. The status of CONTAINMENT CLOSURE is not crucial in this condition since the intact RCS is providing a high pressure barrier to a fission product release. The 60-minute time frame should allow sufficient time to address the temperature increase without a substantial degradation in plant safety.

Finally, in the case where there is an increase in RCS temperature, the RCS is not INTACT or is at reduced inventory, and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because 1) the evaporated reactor coolant may be released directly into the containment atmosphere and subsequently to the environment, and 2) there is reduced reactor coolant inventory above the top of irradiated fuel.

The RCS pressure increase threshold provides a pressure-based indication of RCS heat-up.

Escalation of the emergency classification level would be via IC CS1 or RS1.

CNP Basis Reference(s):

1. CNP Technical Specifications Table 1.1-1
2. 1(2)-OHP-4021-001-004, Plant Cooldown from Hot Standby to Cold Shutdown
3. 1(2)-OHP-4022-017-001, Loss of RHR Cooling
4. NEI 99-01 CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – Loss of Vital DC Power

Initiating Condition: Loss of Vital DC power for 15 minutes or longer

EAL:

CU4.1	**Unusual Event**
< 215 VDC bus voltage indications on Technical Specification **required** 250 VDC vital buses for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

None

Basis:

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. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss. The fifteen minute interval is intended to exclude transient or momentary power losses.

The vital DC buses are the following 250 VDC Class 1E buses (ref. 2, 3):

Train A:	Train B:
1CD (2CD)	1AB (2AB)

There are two, 116 cell, lead-acid storage batteries (1AB (2AB) and 1CD (2CD)) that supplement the output of the battery chargers. They supply DC power to the distribution buses when AC power to the chargers is lost or when transient loads exceed the capacity of the battery chargers (ref. 3).

CNP Technical Specification LCO 3.8.5 requires that one Train A or Train B 250 VDC electrical power subsystem shall be OPERABLE to support one train of the DC Electrical Power Distribution System required by LCO 3.8.10, "Distribution Systems - Shutdown." (ref. 1).

Per SD-DCC-NEEP-104, a 210 VDC lower limit has been identified from the battery service test acceptance criteria. Based on interpolation, the low voltage limit that would provide a 15 minute margin has been determined to be 213 VDC (ref. 4).

An EAL value of 215 VDC has been selected to account for available instrument accuracy. Meter scaling on installed control room instrumentation (10 VDC divisions on a dial indicator) limits the closest value that can be accurately read on the control board to 5 VDC.

This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS7.1.

This IC addresses a loss of vital DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or refueling mode. In these modes, the core decay heat load has been significantly reduced, and coolant system temperatures and pressures are lower; these conditions increase the time available to restore a vital DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, "required" means the vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if Train A is out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an UNUSUAL EVENT. A loss of Vital DC power to Train A would not warrant an emergency classification.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Depending upon the event, escalation of the emergency classification level would be via IC CA1 or CA3, or an IC in Recognition Category R.

CNP Basis Reference(s):

1. Technical Specifications Section 3.8.5 DC Sources - Shutdown
2. UFSAR Figure 8.3-2
3. UFSAR Section 8.3.4 250 Volt DC System (Safety Related)
4. SD-DCC-NEEP-104 250 VDC System
5. NEI 99-01 CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

CU5.1 Unusual Event

Loss of **all** Table C-5 onsite communication methods

OR

Loss of **all** Table C-5 ORO communication methods

OR

Loss of **all** Table C-5 NRC communication methods

Table C-5 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Plant Page	X		
Plant Radios	X	X	
Plant Telephone	X	X	X
ENS Line		X	X
Commercial Telephone		X	X
Microwave Transmission		X	X

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, D – Defueled

Definition(s):

None

Basis:

Onsite/offsite communications include one or more of the systems listed in Table C-5 (ref. 1).

This EAL is the cold condition equivalent of the hot condition EAL SU7.1.

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the State and Berrien County EOCs

The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

CNP Basis Reference(s):

1. CNP Plant Emergency Plan Section F Emergency Communications
2. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting Safety Systems

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

EAL:

CA6.1 Alert

The occurrence of **any** Table C-6 hazardous event

AND EITHER:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Table C-6 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the SEC

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

VISIBLE DAMAGE - Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did **not** cause damage to a structure or any other equipment) is **not** visible damage.

Basis:

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

The first conditional addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

The second conditional addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC CS1 or RS1.

CNP Basis Reference(s):

1. NEI 99-01 CA6

Category H – Hazards and Other Conditions Affecting Plant Safety

EAL Group: ANY (EALs in this category are applicable to any plant condition, hot or cold.)

Hazards are non-plant, system-related events that can directly or indirectly affect plant operation, reactor plant safety or personnel safety.

1. Security

Unauthorized entry attempts into the Protected Area, bomb threats, sabotage attempts, and actual security compromises threatening loss of physical control of the plant.

2. Seismic Event

Natural events such as earthquakes have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety.

3. Natural or Technology Hazard

Other natural and non-naturally occurring events that can cause damage to plant facilities include tornados, FLOODING, hazardous material releases and events restricting site access warranting classification.

4. Fire

Fires can pose significant hazards to personnel and reactor safety. Appropriate for classification are fires within the site Protected Area or which may affect operability of equipment needed for safe shutdown

5. Hazardous Gas

Toxic, corrosive, asphyxiant or flammable gas leaks can affect normal plant operations or preclude access to plant areas required to safely shutdown the plant.

6. Control Room Evacuation

Events that are indicative of loss of Control Room habitability. If the Control Room must be evacuated, additional support for monitoring and controlling plant functions is necessary through the emergency response facilities.

7. SEC 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 SEC the latitude to classify emergency conditions consistent with the established classification criteria based upon SEC judgment.

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1	**Unusual Event**

A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervisor

OR

Notification of a credible security threat directed at the site

OR

A validated notification from the NRC providing information of an aircraft threat

Mode Applicability:

All

Definition(s):

SECURITY CONDITION - Any security event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A security condition does not involve a hostile action.

HOSTILE ACTION - An act toward CNP 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 CNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

Basis:

The security shift supervision is defined as the Security Shift Supervisor.

This EAL is based on the Donald C. Cook Nuclear Plant Security Plan (ref. 1).

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, and HS1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan.*

The first threshold references the Shift Security Supervisor because these are the individuals trained to confirm that a security event is occurring or has occurred (ref. 1). Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.39 information.

The second threshold addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the CNP Plant Security Plan and DBT.

The third threshold addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with Donald C. Cook Nuclear Plant Security Plan.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.

Escalation of the emergency classification level would be via IC HA1.

CNP Basis Reference(s):

1. Donald C. Cook Nuclear Plant Security Plan (Safeguards)
2. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

EAL:

HA1.1 Alert

A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor

OR

A validated notification from NRC of an aircraft attack threat within 30 min. of the site

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward CNP 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 CNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

OWNER CONTROLLED AREA - Area outside the PROTECTED AREA fence that immediately surrounds the plant. Access to this area is generally restricted to those entering on official business.

Basis:

The security shift supervision is defined as the Security Shift Supervisor (ref. 1).

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between the Security Shift Supervisor and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program.*

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The ALERT declaration will also heighten the awareness of Offsite Response

Organizations (OROs), allowing them to be better prepared should it be necessary to consider further actions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

The first threshold is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes any action directed against an ISFSI that is located outside the plant PROTECTED AREA.

The second threshold addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and OROs are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with site-specific security procedures.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may be provided by NORAD through the NRC.

In some cases, it may not be readily apparent if an aircraft impact within the OWNER CONTROLLED AREA was intentional (i.e., a HOSTILE ACTION). It is expected, although not certain, that notification by an appropriate Federal agency to the site would clarify this point. In this case, the appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. The emergency declaration, including one based on other ICs/EALs, should not be unduly delayed while awaiting notification by a Federal agency.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.

CNP Basis Reference(s):

1. Donald C. Cook Nuclear Plant Security Plan (Safeguards)
2. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the plant PROTECTED AREA

EAL:

HS1.1	**Site Area Emergency**
A HOSTILE ACTION is occurring or has occurred within the plant PROTECTED AREA as reported by the Security Shift Supervisor	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward CNP 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 CNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

PROTECTED AREA - The area encompassed by physical barriers to control access to the plant and to the ISFSI.

Basis:

The security shift supervision is defined as the Security Shift Supervisor (ref. 1).

These individuals are the designated onsite personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Donald C. Cook Nuclear Plant Security Plan (Safeguards) information.

This IC addresses the occurrence of a HOSTILE ACTION within the PROTECTED AREA. This event will require rapid response and assistance due to the possibility for damage to plant equipment.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3, 4 5).

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program.*

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The SITE AREA EMERGENCY declaration will mobilize Offsite Response Organization (ORO) resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

This IC does not apply to a HOSTILE ACTION directed at an ISFSI PROTECTED AREA located outside the plant PROTECTED AREA; such an attack should be assessed using IC HA1. It also does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan.

Escalation of the emergency classification level would be via IC FG1.

CNP Basis Reference(s):

1. Donald C. Cook Nuclear Plant Security Plan (Safeguards)
2. NEI 99-01 HS1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE level

EAL:

HU2.1	**Unusual Event**

Control Room personnel feel an actual or potential seismic event

AND

The occurrence of a seismic event is confirmed in manner deemed appropriate by the Shift Manager

Mode Applicability:

All

Definition(s):

None

Basis:

Ground motion acceleration of 0.10g horizontal is the Operating Basis Earthquake for CNP (ref. 1).

To avoid inappropriate emergency classification resulting from spurious actuation of the seismic instrumentation or felt motion not attributable to seismic activity, an offsite agency (USGS, National Earthquake Information Center) can confirm that an earthquake has occurred in the area of the plant. 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 CNP. Alternatively, near real-time seismic activity can be accessed via the NEIC website:

http://earthquake.usgs.gov

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE). An earthquake greater than an OBE but less than a Safe Shutdown Earthquake (SSE) should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., performs walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event (e.g., lateral accelerations in excess of 0.08g). The Shift Manager or SEC may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration (ref. 2).

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

1(2)-OHP-4022-001-007 Earthquake provides the guidance for determining if the OBE earthquake threshold is exceeded and any required response actions. (ref. 2). Because CNP seismic instrumentation does not provide direct and timely indications of having exceeded the OBE ground acceleration, the alternative EAL wording specified in the generic NEI 99-01 HU2 developers note (felt earthquake) is implemented.

CNP Basis Reference(s):

1. FSAR Section 1.3.1 Structures and Equipment
2. 1(2)-OHP-4022-001-007 Earthquake
3. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.1	**Unusual Event**
A tornado strike within the PROTECTED AREA	

Mode Applicability:

All

Definition(s):

PROTECTED AREA - The area encompassed by physical barriers to control access to the plant and to the ISFSI.

Basis:

Response actions associated with a tornado onsite is provided in 12-OHP-4022-001-010 Severe Weather (ref. 1).

If damage is confirmed visually or by other in-plant indications, the event may be escalated to an ALERT under EAL CA6.1 or SA9.1.

A tornado striking (touching down) within the PROTECTED AREA warrants declaration of an UNUSUAL EVENT regardless of the measured wind speed at the meteorological tower. A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

EAL HU3.1 addresses a tornado striking (touching down) within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

CNP Basis Reference(s):

1. 12-OHP-4022-001-010 Severe Weather
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.2	**Unusual Event**
Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode	

Mode Applicability:

All

Definition(s):

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

Refer to EALs CA6.1 or SA9.1 for internal flooding affecting one or more SAFETY SYSTEM trains.

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses FLOODING of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

CNP Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.3	**Unusual Event**
Movement of personnel within the plant PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)	

Mode Applicability:

All

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

PROTECTED AREA - The area encompassed by physical barriers to control access to the plant and to the ISFSI.

Basis:

As used here, the term "offsite" is meant to be areas external to the CNP plant PROTECTED AREA.

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

CNP Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.4	**Unusual Event**
A hazardous event that results in onsite conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)	

Note 7: This EAL does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Mode Applicability:

All

Definition(s):

None

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site FLOODING caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.

This EAL is not intended to apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

CNP Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.1 Unusual Event

A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):

- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

The FIRE is located within **any** Table H-1 area

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Control Room
• Containment
• Auxiliary Building
• Switchgear Areas
• Diesel Generator Rooms
• ESW System enclosures
• AFW Pump Rooms
• Refueling Water Storage Tank
• Condensate Storage Tank

Mode Applicability:

All

Definition(s):

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

Basis:

Table H-1 Fire Areas are based on Fire Hazards Analysis Units No. 1 and 2. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1).

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

For EAL HU4.1 the intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

CNP Basis Reference(s):

1. Fire Hazards Analysis Units No. 1 and 2
2. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.2	**Unusual Event**
Receipt of a single fire alarm (i.e., **no** other indications of a FIRE) **AND** The fire alarm is indicating a FIRE within **any** Table H-1 area **AND** The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Control Room • Containment • Auxiliary Building • Switchgear Areas • Diesel Generator Rooms • ESW System enclosures • AFW Pump Rooms • Refueling Water Storage Tank • Condensate Storage Tank

Mode Applicability:

All

Definition(s):

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

Basis:

Table H-1 Fire Areas are based on Fire Hazards Analysis Units No. 1 and 2. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1).

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

Basis-Related Requirements from Appendix R

(Note: CNP is not an Appendix R plant. This bases is cited only to justify the 30 minute timing component related to a single fire alarm)

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."
>
> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.
>
> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

CNP Basis Reference(s):

1. Fire Hazards Analysis Units No. 1 and 2
2. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.3	**Unusual Event**
A FIRE within the PROTECTED AREA (plant or ISFSI) **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

All

Definition(s):

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

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.

PROTECTED AREA - The area encompassed by physical barriers to control access to the plant and to the ISFSI.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

In addition to a FIRE addressed by EAL HU4.1 or HU4.2, a FIRE within the PROTECTED AREA (plant or ISFSI) not extinguished within 60-minutes may also potentially degrade the level of plant safety. This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

CNP Basis Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.4	**Unusual Event**
A FIRE within the PROTECTED AREA (plant or ISFSI) that requires firefighting support by an offsite fire response agency to extinguish	

Mode Applicability:

All

Definition(s):

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

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.

PROTECTED AREA - The area encompassed by physical barriers to control access to the plant and to the ISFSI.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

If a FIRE within the plant or ISFSI PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

CNP Basis Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gases

Initiating Condition: Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

HA5.1 Alert

Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 rooms or areas

AND

Entry into the room or area is prohibited or IMPEDED (Note 5)

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

Table H-2 Safe Operation & Shutdown Room/Areas

Room/Area	Mode Applicability
Auxiliary Building 573'	4, 5
Auxiliary Building 587' (including D/G room)	1, 2, 3, 4, 5
Auxiliary Building 591'	1, 2, 3, 4
Auxiliary Building 609' (including 4kV room)	1, 2, 3, 4, 5
Auxiliary Building 633'	1, 2, 3, 4
Turbine Building (**All** Levels)	1, 2, 3
Turbine Building 591'	4, 5
Screenhouse	1, 2, 3, 4, 5

Mode Applicability:

All

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

Basis:

If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

The list of plant rooms or areas with entry-related mode applicability identified specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

This IC addresses an event involving a release of a hazardous gas that precludes or IMPEDES access to equipment necessary to maintain normal plant operation, or required for a normal plant cooldown and shutdown. This condition represents an actual or potential substantial degradation of the level of safety of the plant.

An ALERT declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.

Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the SEC's judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

An emergency declaration is not warranted if any of the following conditions apply:

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

This EAL does not apply to firefighting activities that automatically or manually activate a fire suppression system in an area.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

CNP Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Rooms/Areas Tables R-2 & H-2 Bases
2. NEI 99-01 HA5

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations

EAL:

HA6.1	**Alert**
An event has resulted in plant control being transferred from the Control Room to the Local Shutdown Instrumentation	

Mode Applicability:

All

Definition(s):

None

Basis:

Plant control is considered to have been transferred when either 1) control of the plant is no longer maintained in the Control Room or 2) the last licensed operator has left the Control Room, whichever comes first.

The Shift Manager (SM) determines if the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions. 1(2)-OHP-4025-001-001, Emergency Remote Shutdown (ERS), and 1(2)-OHP-4022-CRE-001 Control Room Evacuation provide the instructions for tripping the unit, and maintaining RCS inventory and Hot Shutdown conditions from outside the Control Room (Ref. 1, 2).

This EAL is only applicable when the decision has been made to evacuate the Control Room, not when conditions are being evaluated.

Inability to establish plant control from outside the Control Room escalates this event to a SITE AREA EMERGENCY per EAL HS6.1.

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Escalation of the emergency classification level would be via IC HS6.

CNP Basis Reference(s):

1. 1(2)-OHP-4025-001-001, Emergency Remote Shutdown (ERS)
2. 1(2)-OHP-4022-CRE-001 Control Room Evacuation
3. NEI 99-01 HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

Initiating Condition: Inability to control a key safety function from outside the Control Room

EAL:

HS6.1 Site Area Emergency

An event has resulted in plant control being transferred from the Control Room to the Local Shutdown Instrumentation

AND

Control of **any** of the following key safety functions is **not** reestablished within 15 min. (Note 1):

- Reactivity Control (modes 1, 2 and 3 **only**)
- Core Cooling
- RCS heat removal

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown, 5 – Cold Shutdown, 6 - Refueling

Definition(s):

None

Basis:

The 15-minute time period starts when either 1) control of the plant is no longer maintained in the Control Room or 2) the last licensed operator has left the Control Room, whichever comes first.

The Shift Manager determines if the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions. 1(2)-OHP-4025-001-001, Emergency Remote Shutdown (ERS) and 1(2)-OHP-4022-CRE-001 Control Room Evacuation provides the instructions for tripping the unit, and maintaining RCS inventory and Hot Shutdown conditions from outside the Control Room (ref. 1, 2).

The intent of this EAL is to capture events in which control of the plant cannot be reestablished in a timely manner. The fifteen minute time for transfer starts when the last licensed operator has left the Control Room (not when 1(2)-OHP-4025-001-001 or 1(2)-OHP-4022-CRE-001 is entered). The time interval is based on how quickly control must be reestablished without core uncovery and/or core damage. The determination of whether or not control is established from outside the Control Room is based on SEC judgment. The SEC is expected to make a reasonable, informed judgment that control of the plant from outside the Control Room cannot be established within the fifteen minute interval.

Once the Control Room is evacuated, the objective is to establish control of important plant equipment and maintain knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on 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), RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink).

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on SEC judgment. The SEC is expected to make a reasonable, informed judgment within 15 minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

Escalation of the emergency classification level would be via IC FG1 or CG1.

CNP Basis Reference(s):

1. 1(2)-OHP-4025-001-001, Emergency Remote Shutdown (ERS)
2. 1(2)-OHP-4022-CRE-001 Control Room Evacuation
3. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

Initiating Condition: Other conditions existing that in the judgment of the SEC warrant declaration of a UE

EAL:

HU7.1	**Unusual Event**
Other conditions exist which in the judgment of the SEC indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.	

Mode Applicability:

All

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

The SEC is the designated onsite individual having the responsibility and authority for implementing the CNP Emergency Plan (ref. 1). The Shift Manager (SM) initially acts in the capacity of the SEC and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the SEC, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the SEC to fall under the emergency classification level description for an UNUSUAL EVENT.

CNP Basis Reference(s):

1. CNP Emergency Plan section B.5.a.1 Site Emergency Coordinator
2. CNP Emergency Plan section B.1.k On-Shift Operations Personnel
3. NEI 99-01 HU7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

Initiating Condition: Other conditions exist that in the judgment of the SEC warrant declaration of an ALERT

EAL:

HA7.1 Alert

Other conditions exist which, in the judgment of the SEC, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward CNP 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 CNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

Basis:

The SEC is the designated onsite individual having the responsibility and authority for implementing the CNP Emergency Plan (ref. 1). The Shift Manager (SM) initially acts in the capacity of the SEC and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the SEC, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the SEC to fall under the emergency classification level description for an ALERT.

CNP Basis Reference(s):

1. CNP Emergency Plan section B.5.a.1 Site Emergency Coordinator
2. CNP Emergency Plan section B.1.k On-Shift Operations Personnel
3. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

Initiating Condition: Other conditions existing that in the judgment of the SEC warrant declaration of a SITE AREA EMERGENCY

EAL:

HS7.1 **Site Area Emergency**
Other conditions exist which in the judgment of the SEC indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward CNP 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 CNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area)

Basis:

The SEC is the designated onsite individual having the responsibility and authority for implementing the CNP Emergency Plan (ref. 1). The Shift Manager (SM) initially acts in the capacity of the SEC and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the SEC, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the SEC to fall under the emergency classification level description for a SITE AREA EMERGENCY.

CNP Basis Reference(s):

1. CNP Emergency Plan section B.5.a.1 Site Emergency Coordinator
2. CNP Emergency Plan section B.1.k On-Shift Operations Personnel
3. NEI 99-01 HS7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

Initiating Condition: Other conditions exist which in the judgment of the SEC warrant declaration of a GENERAL EMERGENCY

EAL:

HG7.1	**General Emergency**
Other conditions exist which in the judgment of the SEC indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward CNP 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 CNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

IMMINENT - The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

Basis:

The SEC is the designated onsite individual having the responsibility and authority for implementing the CNP Emergency Plan (ref. 1). The Shift Manager (SM) initially acts in the capacity of the SEC and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the SEC, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

Releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the site boundary.

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the SEC to fall under the emergency classification level description for a GENERAL EMERGENCY.

CNP Basis Reference(s):

1. CNP Emergency Plan section B.5.a.1 Site Emergency Coordinator
2. CNP Emergency Plan section B.1.k On-Shift Operations Personnel
3. NEI 99-01 HG7

Category S – System Malfunction

EAL Group: Hot Conditions (RCS temperature > 200°F); EALs in this category are applicable only in one or more hot operating modes.

Numerous system-related equipment failure events that warrant emergency classification have been identified in this category. They may pose actual or potential threats to plant safety.

The events of this category pertain to the following subcategories:

1. Loss of Emergency AC Power

Loss of emergency electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite sources for 4.16KV AC emergency buses.

2. Loss of Vital DC Power

Loss of emergency electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of vital plant 250 VDC power sources.

3. Loss of Control Room Indications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant increase from these base-line levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

5. RCS Leakage

The reactor vessel provides a volume for the coolant that covers the reactor core. The reactor pressure vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail. Excessive RCS leakage greater than Technical Specification limits indicates potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and containment integrity.

6. RPS Failure

This subcategory includes events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RPS to complete a reactor trip comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean any 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.

7. Loss of Communications

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

8. Containment Failure

Failure of containment isolation capability (under conditions in which the containment is not currently challenged) warrants emergency classification. Failure of containment pressure control capability also warrants emergency classification.

9. Hazardous Event Affecting Safety Systems

Various natural and technological events that result in degraded plant safety system performance or significant VISIBLE DAMAGE warrant emergency classification under this subcategory.

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite AC power capability to emergency buses for 15 minutes or longer

EAL:

SU1.1	**Unusual Event**
Loss of **all** offsite AC power capability, Table S-1, to emergency 4.16KV buses T11A (T21A) and T11D (T21D) for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 AC Power Sources
Offsite:
• Unit Auxiliary Xmr TR1AB (TR2AB)
• Unit Auxiliary Xmr TR1CD (TR2CD)
• Reserve Auxiliary Xmr TR101AB (TR201AB)
• Reserve Auxiliary Xmr TR101CD (TR201CD)
• 69/4.16 kV Alternate Xmr TR12EP-1
Onsite:
• EDG 1AB (2AB)
• EDG 1CD (2CD)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

Basis:

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the emergency buses, whether or not the buses are powered from it.

A list of offsite AC power sources credited for this EAL are specified in Table S-1.

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. 4.16KV buses T11A (T21A) and T11D (T21D) are the emergency (essential) buses (ref. 1). While generating, auxiliary power is normally supplied from the generator terminals through the unit auxiliary transformers (TR1AB and TR1CD for Unit 1 and TR2AB and TR2CD for Unit 2). When the plant trips or the plant is shutdown the station auxiliaries are transferred to the preferred offsite power source (that is, to reserve auxiliary transformers TR101AB and

TR101CD for Unit 1 and TR201AB and TR201CD for Unit 2) to assure continued power to equipment when the main generator is off-line (ref. 1, 2, 3).

In addition, an alternate offsite power source, a 69/4.16kV transformer (TR12EP-1), located at the plant site, has the necessary capacity to operate one train of the engineered safeguard equipment in one unit while supplying one train of the safe shutdown power in the other.

T11A (T21A) and T11D (T21D) also each have an emergency diesel generator which supply onsite electrical power to the bus automatically in the event that the preferred offsite sources become unavailable (ref. 1, 2, 3).

The Supplemental Diesel Generators (SDGs) are not credited as an AC power source for this EAL.

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC emergency buses. This condition represents a potential reduction in the level of safety of the plant.

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the emergency buses, whether or not the buses are powered from it.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of offsite power.

Escalation of the emergency classification level would be via IC SA1.

CNP Basis Reference(s):

1. UFSAR Figure 8.1-1A(B) Main Auxiliary One-Line Diagram
2. UFSAR Section 8.0 Electrical Systems
3. 1(2)-OHP-4022-001-005 Loss of Offsite Power with Reactor Shutdown
4. NEI 99-01 SU1

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all but one** AC power source to emergency buses for 15 minutes or longer

EAL:

SA1.1 Alert

AC power capability, Table S-1, to emergency 4.16KV buses T11A (T21A) and T11D (T21D) reduced to a single power source for ≥ 15 min. (Note 1)

AND

Any additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 AC Power Sources
Offsite:
• Unit Auxiliary Xmr TR1AB (TR2AB)
• Unit Auxiliary Xmr TR1CD (TR2CD)
• Reserve Auxiliary Xmr TR101AB (TR201AB)
• Reserve Auxiliary Xmr TR101CD (TR201CD)
• 69/4.16 kV Alternate Xmr TR12EP-1
Onsite:
• EDG 1AB (2AB)
• EDG 1CD (2CD)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the emergency buses, whether or not the buses are powered from it.

A list of onsite and offsite AC power sources credited for this EAL are specified in Table S-1.

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. 4.16KV buses T11A (T21A) and T11D (T21D) are the emergency (essential) buses (ref. 1). While generating, auxiliary power is normally supplied from the generator terminals through the unit auxiliary transformers (TR1AB and TR1CD for Unit 1 and TR2AB and TR2CD for Unit 2). When the plant trips or the plant is shutdown the station auxiliaries are transferred to the preferred offsite power source (that is, to reserve auxiliary transformers TR101AB and TR101CD for Unit 1 and TR201AB and TR201CD for Unit 2) to assure continued power to equipment when the main generator is off-line (ref. 1, 2, 3).

In addition, an alternate offsite power source, a 69/4.16kV transformer (TR12EP-1), located at the plant site, has the necessary capacity to operate one train of the engineered safeguard equipment in one unit while supplying one train of the safe shutdown power in the other.

T11A (T21A) and T11D (T21D) also each have an emergency diesel generator which supply onsite electrical power to the bus automatically in the event that the preferred offsite sources become unavailable (ref. 1, 2, 3).

The Supplemental Diesel Generators (SDGs) are not credited as an AC power source for this EAL.

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an emergency bus. Examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being fed from an offsite power source.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

Escalation of the emergency classification level would be via IC SS1.

CNP Basis Reference(s):

1. UFSAR Figure 8.1-1A(B) Main Auxiliary One-Line Diagram
2. UFSAR Section 8.0 Electrical Systems
3. 1(2)-OHP-4022-001-005 Loss of Offsite Power with Reactor Shutdown
4. NEI 99-01 SA1

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite power and **all** onsite AC power to emergency buses for 15 minutes or longer

EAL:

SS1.1	**Site Area Emergency**
Loss of **all** offsite and **all** onsite AC power to emergency 4.16KV buses T11A (T21A) and T11D (T21D) for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. 4.16KV buses T11A (T21A) and T11D (T21D) are the emergency (essential) buses (ref. 1). While generating, auxiliary power is normally supplied from the generator terminals through the unit auxiliary transformers (TR1AB and TR1CD for Unit 1 and TR2AB and TR2CD for Unit 2). When the plant trips or the plant is shutdown the station auxiliaries are transferred to the preferred offsite power source (that is, to reserve auxiliary transformers TR101AB and TR101CD for Unit 1 and TR201AB and TR201CD for Unit 2) to assure continued power to equipment when the main generator is off-line (ref. 1, 2, 3).

In addition, an alternate offsite power source, a 69/4.16kV transformer (TR12EP-1), located at the plant site, has the necessary capacity to operate one train of the engineered safeguard equipment in one unit while supplying one train of the safe shutdown power in the other.

T11A (T21A) and T11D (T21D) also each have an emergency diesel generator which supply onsite electrical power to the bus automatically in the event that the preferred offsite sources become unavailable (ref. 1, 2, 3).

The Supplemental Diesel Generators (SDGs) or any other alternative AC power source capable of powering an emergency bus can also be credited as an AC power source for this EAL.

The 15-minute interval begins when both offsite and onsite AC power capability are lost (ref. 4).

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

CNP Basis Reference(s):

1. UFSAR Figure 8.1-1A(B) Main Auxiliary One-Line Diagram
2. UFSAR Section 8.0 Electrical Systems
3. 1(2)-OHP-4022-001-005 Loss of Offsite Power with Reactor Shutdown
4. 1(2)-OHP-4023-ECA-0.0 Loss of All AC Power
5. NEI 99-01 SS1

Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses

EAL:

SG1.1 General Emergency
Loss of **all** offsite and **all** onsite AC power to emergency 4.16KV buses T11A (T21A) and T11D (T21D)
AND EITHER:
• Restoration of at least one emergency bus in < 4 hours is **not** likely (Note 1)
• CSFST Core Cooling RED Path (F.0-2) conditions met

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

This EAL is indicated by the extended loss of all offsite and onsite AC power capability to 4.16KV emergency buses T11A (T21A) and T11D (T21D) either for greater then the CNP Station Blackout (SBO) coping analysis time (4 hrs.) (ref. 1, 4) or that has resulted in indications of an actual loss of adequate core cooling.

Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED PATH conditions being met. (ref. 5).

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. 4.16KV buses T11A (T21A) and T11D (T21D) are the emergency (essential) buses (ref. 1). While generating, auxiliary power is normally supplied from the generator terminals through the unit auxiliary transformers (TR1AB and TR1CD for Unit 1 and TR2AB and TR2CD for Unit 2). When the plant trips or the plant is shutdown the station auxiliaries are transferred to the preferred offsite power source (that is, to reserve auxiliary transformers TR101AB and TR101CD for Unit 1 and TR201AB and TR201CD for Unit 2) to assure continued power to equipment when the main generator is off-line (ref. 1, 2, 3).

In addition, an alternate offsite power source, a 69/4.16kV transformer (TR12EP-1), located at the plant site, has the necessary capacity to operate one train of the engineered safeguard equipment in one unit while supplying one train of the safe shutdown power in the other.

T11A (T21A) and T11D (T21D) also each have an emergency diesel generator which supply onsite electrical power to the bus automatically in the event that the preferred offsite sources become unavailable (ref. 1, 2, 3).

The Supplemental Diesel Generators (SDGs) or any other alternative AC power source capable of powering an emergency bus can also be credited as an AC power source for this EAL.

Four hours is the station blackout coping time (ref. 4).

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on SEC judgment as it relates to imminent Loss of fission product barriers and degraded ability to monitor fission product barriers. Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED PATH conditions being met. Specifically, Core Cooling RED PATH conditions exist if either the five highest core exit TCs are reading greater than or equal to 1200°F or core exit TCs are reading greater than or equal to 757°F with RCS subcooling less than or equal to 40°F, and RVLIS indication is less than or equal to that specified based on the number of RCPs running (ref. 5).

This IC addresses a prolonged loss of all power sources to AC emergency buses. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A prolonged loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

The EAL should require declaration of a GENERAL EMERGENCY prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from SITE AREA EMERGENCY will occur if it is projected that power cannot be restored to at least one AC emergency bus by the end of the analyzed station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of challenges to multiple fission product barriers.

The estimate for restoring at least one emergency bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, protective actions for the public.

The EAL will also require a GENERAL EMERGENCY declaration if the loss of AC power results in parameters that indicate an inability to adequately remove decay heat from the core.

CNP Basis Reference(s):

1. UFSAR Figure 8.1-1A(B) Main Auxiliary One-Line Diagram
2. UFSAR Section 8.0 Electrical Systems
3. 1(2)-OHP-4023-ECA-0.0 Loss of All AC Power
4. UFSAR Section 8.7 Station Blackout
5. 1(2)-OHP-4023-F-0.2 Core Cooling
6. NEI 99-01 SG1

Category: S – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

Initiating Condition: Loss of **all** vital DC power for 15 minutes or longer

EAL:

SS2.1	**Site Area Emergency**
Loss of **all** 250 VDC power based on bus voltage indications < 215 VDC on **all** vital DC buses 1CD (2CD) (Train A) and 1AB (2AB) (Train B) for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

The vital DC buses are the following 250 VDC Class 1E buses (ref. 1, 2, 3):

Train A:	Train B:
1CD (2CD)	1AB (2AB)

There are two, 116 cell, lead-acid storage batteries (1AB (2AB) and 1CD (2CD)) that supplement the output of the battery chargers. They supply DC power to the distribution buses when AC power to the chargers is lost or when transient loads exceed the capacity of the battery chargers (ref. 3).

CNP Technical Specification LCO 3.8.4 requires that both Train A and Train B 250 VDC electrical power subsystem shall be OPERABLE to support both trains of the DC Electrical Power Distribution System required by LCO 3.8.9, "Distribution Systems - Operating." (ref. 1).

Per SD-DCC-NEEP-104, a 210 VDC lower limit has been identified from the battery service test acceptance criteria. Based on interpolation, the low voltage limit that would provide a 15 minute margin has been determined to be 213 VDC (ref. 4).

An EAL value of 215 VDC has been selected to account for available instrument accuracy. Meter scaling on installed control room instrumentation (10 VDC divisions on a dial indicator) limits the closest value that can be accurately read on the control board to 5 VDC.

This IC addresses a loss of vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG2.

CNP Basis Reference(s):

1. Technical Specifications Section 3.8.4 DC Sources - Operating
2. UFSAR Figure 8.3-2
3. UFSAR Section 8.3.4 250 Volt DC System (Safety Related)
4. SD-DCC-NEEP-104 250 VDC System
5. NEI 99-01 SS8

Category: S –System Malfunction

Subcategory: 2 – Loss of Vital DC Power

Initiating Condition: Loss of **all** AC and vital DC power sources for 15 minutes or longer

EAL:

SG2.1 General Emergency
Loss of **all** offsite and **all** onsite AC power to emergency 4.16KV buses T11A (T21A) and T11D (T21D) for ≥ 15 min.
AND
Loss of **all** 250 VDC power based on bus voltage indications < 215 VDC on **all** vital DC buses 1CD (2CD) (Train A) and 1AB (2AB) (Train B) for ≥ 15 min.
(Note 1)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

This EAL is indicated by the loss of all offsite and onsite emergency AC power capability to 4.16KV emergency buses T11A (T21A) and T11D (T21D) for greater than 15 minutes in combination with degraded vital DC power voltage. This EAL addresses operating experience from the March 2011 accident at Fukushima Daiichi.

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. 4.16KV buses T11A (T21A) and T11D (T21D) are the emergency (essential) buses (ref. 1). While generating, auxiliary power is normally supplied from the generator terminals through the unit auxiliary transformers (TR1AB and TR1CD for Unit 1 and TR2AB and TR2CD for Unit 2). When the plant trips or the plant is shutdown the station auxiliaries are transferred to the preferred offsite power source (that is, to reserve auxiliary transformers TR101AB and TR101CD for Unit 1 and TR201AB and TR201CD for Unit 2) to assure continued power to equipment when the main generator is off-line (ref. 1, 2, 3).

In addition, an alternate offsite power source, a 69/4.16kV transformer (TR12EP-1), located at the plant site, has the necessary capacity to operate one train of the engineered safeguard equipment in one unit while supplying one train of the safe shutdown power in the other.

T11A (T21A) and T11D (T21D) also each have an emergency diesel generator which supply onsite electrical power to the bus automatically in the event that the preferred offsite sources become unavailable (ref. 1, 2, 3).

The Supplemental Diesel Generators (SDGs) or any other alternative AC power source capable of powering an emergency bus can also be credited as an AC power source for this EAL.

The vital DC buses are the following 250 VDC Class 1E buses (ref. 4, 5, 6):

Train A:	Train B:
1CD (2CD)	1AB (2AB)

There are two, 116 cell, lead-acid storage batteries (1AB (2AB) and 1CD (2CD)) that supplement the output of the battery chargers. They supply DC power to the distribution buses when AC power to the chargers is lost or when transient loads exceed the capacity of the battery chargers (ref. 6).

CNP Technical Specification LCO 3.8.4 requires that both Train A and Train B 250 VDC electrical power subsystem shall be OPERABLE to support both trains of the DC Electrical Power Distribution System required by LCO 3.8.9, "Distribution Systems - Operating." (ref. 4).

Per SD-DCC-NEEP-104, a 210 VDC lower limit has been identified from the battery service test acceptance criteria. Based on interpolation, the low voltage limit that would provide a 15 minute margin has been determined to be 213 VDC (ref. 7).

An EAL value of 215 VDC has been selected to account for available instrument accuracy. Meter scaling on installed control room instrumentation (10 VDC divisions on a dial indicator) limits the closest value that can be accurately read on the control board to 5 VDC.

This IC addresses a concurrent and prolonged loss of both emergency AC and Vital DC power. A loss of all emergency AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both emergency AC and vital DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

CNP Basis Reference(s):

1. UFSAR Figure 8.1-1A(B) Main Auxiliary One-Line Diagram
2. UFSAR Section 8.0 Electrical Systems
3. 1(2)-OHP-4023-ECA-0.0 Loss of All AC Power
4. Technical Specifications Section 3.8.4 DC Sources - Operating
5. UFSAR Figure 8.3-2

6. UFSAR Section 8.3.4 250 Volt DC System (Safety Related)
7. SD-DCC-NEEP-104 250 VDC System
8. NEI 99-01 SG8

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer

EAL:

SU3.1	**Unusual Event**
An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-2 Safety System Parameters
• Reactor power
• RCS level
• RCS pressure
• Core Exit TC temperature
• Level in at least one SG
• Auxiliary feed flow in at least one SG

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Process Computer, which displays SPDS required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1).

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via IC SA3.

CNP Basis Reference(s):

1. UFSAR Section 7.5 Engineered Safety Features Instrumentation
2. NEI 99-01 SU2

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress

EAL:

SA3.1 Alert

An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

Any significant transient is in progress, Table S-3

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core Exit TC temperature • Level in at least one SG • Auxiliary feed flow in at least one SG

Table S-3 Significant Transients
• Reactor trip • Runback ≥ 25% thermal power • Electrical load rejection > 25% of full electrical load • ECCS actuation

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Computer, which displays SPDS required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1).

Significant transients are listed in Table S-3 and include response to automatic or manually initiated functions such as reactor trips, runbacks involving greater than or equal to 25% thermal power change, electrical load rejections of greater than 25% full electrical load or ECCS (SI) injection actuations.

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via ICs FS1 or IC RS1

CNP Basis Reference(s):

1. UFSAR Section 7.5 Engineered Safety Features Instrumentation
2. NEI 99-01 SA2

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.1	**Unusual Event**
Sample analysis indicates RCS activity > Technical Specification Section 3.4.16 limits	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

CNP Basis Reference(s):

1. CNP Technical Specifications section 3.4.16 RCS Specific Activity
2. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1 Unusual Event

RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min.

OR

RCS identified leakage > 25 gpm for ≥ 15 min.

OR

Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min. (Note 1)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Manual or computer-based methods of performing an RCS inventory balance are normally used to determine RCS leakage (ref. 1).

Identified leakage includes

- Leakage such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank, or
- Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage, or
- RCS leakage through a steam generator to the secondary system (ref. 2).

Unidentified leakage is all leakage (except RCP seal water injection or leakoff) that is not identified leakage (ref. 2).

Pressure Boundary leakage is leakage (except SG leakage) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall (ref. 2)

RCS leakage outside of the containment that is not considered identified or unidentified leakage per Technical Specifications includes leakage via interfacing systems such as RCS to the Component Cooling Water, or systems that directly see RCS pressure outside containment such as Chemical & Volume Control System, Nuclear Sampling system and Residual Heat Removal system (when in the shutdown cooling mode) (ref. 3, 4)

Escalation of this EAL to the ALERT level is via Category F, Fission Product Barrier Degradation, EAL FA1.1.

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

The first and second EAL conditions are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications). The third condition addresses an RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These conditions thus apply to leakage into the containment, a secondary-side system (e.g., steam generator tube leakage) or a location outside of containment.

The leak rate values for each condition were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). The first condition uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. An emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated).

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

Escalation of the emergency classification level would be via ICs of Recognition Category R or F.

CNP Basis Reference(s):

1. 1(2)-OHP-4030-102-016 Reactor Coolant System Leak Test
2. CNP Technical Specifications Definitions section 1.1
3. UFSAR Section 4.2.7 Leakage
4. 1(2)-OHP-4022-002-020 Excessive Reactor Coolant Leakage
5. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

SU6.1	**Unusual Event**

An automatic trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** RPS setpoint is exceeded

AND

A subsequent automatic trip or manual trip action taken at the reactor control console (reactor trip switches) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) trip function. A reactor trip is automatically initiated by the RPS when certain continuously monitored parameters exceed predetermined setpoints (ref. 1, 2).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. 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 and a lowering of power into the source range. A successful trip has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2, 3, 4).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the reactor control console; manual reactor trip switches. Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as tripping the main turbine, locally opening reactor trip breakers, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 4).

Following any automatic RPS trip signal, E-0 (ref. 2) and /FR-S.1 (ref. 4) prescribe insertion of redundant manual trip signals to back up the automatic RPS trip function and ensure reactor shutdown is achieved. Even if the first subsequent manual trip signal inserts all control rods to the full-in position immediately after the initial failure of the automatic trip, the lowest level of classification that must be declared is an UNUSUAL EVENT (ref. 4).

A reactor trip resulting from actuation of the ATWS Mitigation System Actuation Circuitry (AMSAC) logic that results in full insertion of control rods and diminishing neutron flux is considered a successful reactor trip. AMSAC automatically initiates auxiliary feedwater and a turbine trip under conditions indicative of an Anticipated Transient Without Scram (ATWS) event (ref. 5).

In the event that the operator identifies a reactor trip is imminent and initiates a successful manual reactor trip before the automatic RPS trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. However, if subsequent manual reactor trip actions fail to reduce reactor power below 5%, the event escalates to the ALERT under EAL SA6.1.

If by procedure, operator actions include the initiation of an immediate manual trip following receipt of an automatic trip signal and there are no clear indications that the automatic trip failed (such as a time delay following indications that a trip setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic trip or manual actions. If a subsequent review of the trip actuation indications reveals that the automatic trip did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic trip is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor trip, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip) using a different switch. Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an ALERT via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an UNUSUAL EVENT declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor trip signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the trip failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

CNP Basis Reference(s):

1. CNP Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. 1(2)-OHP04023-E-0 Reactor Trip or Safety Injection
3. 1(2)-OHP04023-F-0.1 Critical Safety Function Status Trees - Subcriticality
4. 1(2)-OHP-4023-FR-S-1 Response to Nuclear Power Generation/ATWS
5. UFSAR Section 3.3.3 Anticipated Transients Without Scram

6 NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

SU6.2 Unusual Event

A manual trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** manual trip action was initiated

AND

A subsequent automatic trip or manual trip action taken at the reactor control console (reactor trip switches) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

This EAL addresses a failure of a manually initiated trip in the absence of having exceeded an automatic RPS trip setpoint and a subsequent automatic or manual trip is successful in shutting down the reactor (reactor power < 5%). (ref. 1).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. 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 and a lowering of power into the source range. A successful trip has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2, 3, 4).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the reactor control console; manual reactor trip switches. Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as tripping the main turbine, locally opening reactor trip breakers, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 4).

Following the failure of any manual trip signal, E-0 (ref. 2) and FR-S.1 (ref. 4) prescribe insertion of redundant manual trip signals to back up the RPS trip function and ensure reactor shutdown is achieved. Even if a subsequent automatic trip signal or the first subsequent manual trip signal inserts all control rods to the full-in position immediately after the initial failure of the manual trip, the lowest level of classification that must be declared is an UNUSUAL EVENT (ref. 4).

A reactor trip resulting from actuation of the ATWS Mitigation System Actuation Circuitry (AMSAC) logic that results in full insertion of control rods and diminishing neutron flux is considered a successful reactor trip. AMSAC automatically initiates auxiliary feedwater and a turbine trip under conditions indicative of an Anticipated Transient Without Scram (ATWS) event (ref. 5).

If both subsequent automatic and subsequent manual reactor trip actions in the Control Room fail to reduce reactor power below the power associated with the safety system design (< 5%) following a failure of an initial manual trip, the event escalates to an ALERT under EAL SA6.1.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic trip is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor trip, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip) using a different switch. Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an ALERT via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an UNUSUAL EVENT declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor trip signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the trip failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

CNP Basis Reference(s):

1. CNP Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. 1(2)-OHP04023-E-0 Reactor Trip or Safety Injection
3. 1(2)-OHP04023-F-0.1 Critical Safety Function Status Trees - Subcriticality
4. 1(2)-OHP-4023-FR-S-1 Response to Nuclear Power Generation/ATWS
5. UFSAR Section 3.3.3 Anticipated Transients Without Scram
6. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control console are not successful in shutting down the reactor

EAL:

SA6.1 Alert

An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%

AND

Manual trip actions taken at the reactor control console (reactor trip switches) are **not** successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

This EAL addresses any automatic or manual reactor trip signal that fails to shut down the reactor (reactor power < 5%) followed by a subsequent manual trip that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed (ref. 1, 2).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the reactor control console; manual reactor trip switches. Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as tripping the main turbine, locally opening reactor trip breakers, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 4).

A reactor trip resulting from actuation of the ATWS Mitigation System Actuation Circuitry (AMSAC) logic that results in full insertion of control rods and diminishing neutron flux is considered a successful reactor trip. AMSAC automatically initiates auxiliary feedwater and a turbine trip under conditions indicative of an Anticipated Transient Without Scram (ATWS) event (ref. 5).

5% rated power is a minimum reading on the power range scale that indicates continued power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below 5%, plant response will be similar to that observed during a

normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than 5 % power (ref. 3, 4).

Escalation of this event to a SITE AREA EMERGENCY would be under EAL SS6.1 or SEC judgment.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control console (e.g., locally opening breakers). Actions taken at backpanels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control console".

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shut down the reactor is prolonged enough to cause a challenge to the core cooling or RCS heat removal safety functions, the emergency classification level will escalate to a SITE AREA EMERGENCY via IC SS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6 or FS1, an ALERT declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an ALERT declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

CNP Basis Reference(s):

1. CNP Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. 1(2)-OHP04023-E-0 Reactor Trip or Safety Injection
3. 1(2)-OHP04023-F-0.1 Critical Safety Function Status Trees - Subcriticality
4. 1(2)-OHP-4023-FR-S-1 Response to Nuclear Power Generation/ATWS
5. UFSAR Section 3.3.3 Anticipated Transients Without Scram
6. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%

AND

All actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5%

AND EITHER:

- CSFST Core Cooling RED Path (F-0.2) conditions met
- CSFST Heat Sink RED Path (F-0.3) conditions met

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

This EAL addresses the following:

- Any automatic reactor trip signal followed by a manual trip that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed (EAL SA6.1), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

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.

Reactor shutdown achieved by use of FR-S.1 Response to Nuclear Power Generation/ATWS such as tripping the main turbine, locally opening reactor trip breakers, emergency boration or manually driving control rods are also credited as a successful manual trip provided reactor power can be reduced below 5% before indications of an extreme challenge to either core cooling or heat removal exist (ref. 1, 2).

5% rated power is a minimum reading on the power range scale that indicates continued power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below 5%, plant response will be similar to that observed during a

normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than 5 % power (ref. 1, 2).

Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED PATH conditions being met. Specifically, Core Cooling RED PATH conditions exist if either the five highest core exit TCs are reading greater than or equal to 1200°F or core exit TCs are reading greater than or equal to 757°F with RCS subcooling less than or equal 40°F and RVLIS level less than or equal to that specified based on the number of RCPs running (ref. 3).

Indication of inability to adequately remove heat from the RCS is manifested by CSFST Heat Sink RED PATH conditions being met (ref. 2). Specifically, Heat Sink RED PATH conditions exist if narrow range level in at least one steam generator is not greater than 13% (28% Adverse Containment Conditions) and total feedwater flow to the steam generators is less than or equal to 240,000 lbm/hr. (ref. 4).

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a SITE AREA EMERGENCY.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shut down the reactor. The inclusion of this IC and EAL ensures the timely declaration of a SITE AREA EMERGENCY in response to prolonged failure to shutdown the reactor.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Escalation of the emergency classification level would be via IC RG1 or FG1.

CNP Basis Reference(s):

1. 1(2)-OHP04023-F-0.1 Critical Safety Function Status Trees - Subcriticality
2. 1(2)-OHP-4023-FR-S-1 Response to Nuclear Power Generation/ATWS
3. 1(2)-OHP04023-F-0.2 Critical Safety Function Status Trees – Core Cooling
4. 1(2)-OHP04023-F-0.3 Critical Safety Function Status Trees – Heat Sink
5. NEI 99-01 SS5

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

SU7.1 Unusual Event

Loss of **all** Table S-4 onsite communication methods

OR

Loss of **all** Table S-4 ORO communication methods

OR

Loss of **all** Table S-4 NRC communication methods

Table S-4 Communication Methods

System	Onsite	ORO	NRC
Plant Page	X		
Plant Radios	X	X	
Plant Telephone	X	X	X
ENS Line		X	X
Commercial Telephone		X	X
Microwave Transmission		X	X

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Onsite/offsite communications include one or more of the systems listed in Table C-5 (ref. 1).

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of onsite information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the State and Berrien County EOCs

The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

CNP Basis Reference(s):

1. CNP Plant Emergency Plan Section F Emergency Communications
2. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 8 – Containment Failure

Initiating Condition: Failure to isolate containment or loss of containment pressure control.

EAL:

SU8.1 Unusual Event

Any penetration is **not** isolated within 15 min. of a VALID containment isolation signal

OR

Containment pressure > 2.8 psig with < one full train of containment depressurization equipment operating per design for ≥ 15 min. (Note 9)

(Note 1)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 9: One Containment Spray System train and one Containment Air Recirculation Fan comprise one full train of depressurization equipment.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

VALID - An indication, report, or condition, is considered to be valid when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

Basis:

The containment isolation system provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the outside environment in the event of a design basis accident (ref. 1).

Containment pressure control is achieved through the Containment Spray System and the Containment Air Recirculation/Hydrogen Skimmer System. Failure of either of these systems may allow steam to build up within containment, and, unabated, this steam buildup may cause the internal containment pressure buildup to exceed the design pressure of 12 psig. Studies have shown that the containment can withstand pressures well above this value.

Both the recirculation fans and the containment spray pumps are actuated automatically (time delayed) following receipt of a HI or HI HI (Phase B) containment pressure signal, respectively. Since the HI HI containment pressure setpoint is less than or equal to 2.8 PSI, then greater than 2.8 PSI would be the containment pressure greater than the setpoint at which the equipment was supposed to have actuated per design. If these systems should fail to start automatically per design, a successful manual start within 15 minutes would preclude exceeding this Containment Potential Loss threshold. (ref. 2, 3, 4)

This EAL addresses a failure of one or more containment penetrations to automatically isolate (close) when required by an actuation signal. It also addresses an event that results in high containment pressure with a concurrent failure of containment pressure control systems. Absent challenges to another fission product barrier, either condition represents potential degradation of the level of safety of the plant.

For the first condition, the containment isolation signal must be generated as the result of an off-normal/accident condition (e.g., a safety injection or high containment pressure); a failure resulting from testing or maintenance does not warrant classification. The determination of containment and penetration status – isolated or not isolated – should be made in accordance with the appropriate criteria contained in the plant AOPs and EOPs. This condition includes the failure of Containment Ventilation Isolation to actuate on a VALID signal. The 15-minute criterion is included to allow operators time to manually isolate the required penetrations, if possible.

The second condition addresses a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment sprays or containment recirculation fans) are either lost or performing in a degraded manner.

This event would escalate to a SITE AREA EMERGENCY in accordance with IC FS1 if there were a concurrent loss or potential loss of either the Fuel Clad or RCS fission product barriers.

CNP Basis Reference(s):

1. UFSAR Section 5.4 Containment Isolation System
2. UFSAR Section 5.5.3 System Description
3. UFSAR Section 6.3 Containment Spray Systems
4. EC-0000052930 Unit 1 Return to Normal Operating Pressure and Temperature (NOP/NOT)
5. NEI 99-01 SU7

Category: S – System Malfunction

Subcategory: 9 – Hazardous Event Affecting Safety Systems

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

EAL:

SA9.1 Alert

The occurrence of **any** Table S-5 hazardous event

AND EITHER:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Table S-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the SEC

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

VISIBLE DAMAGE - Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure. Damage resulting from an equipment failure and limited to the failed component (i.e., the failure did **not** cause damage to a structure or any other equipment) is **not** visible damage.

Basis:

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

The first condition addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

The second condition addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC FS1 or RS1.

CNP Basis Reference(s):

1. NEI 99-01 SA9

Category F – Fission Product Barrier Degradation

EAL Group: Hot Conditions (RCS temperature > 200°F); EALs in this category are applicable only in one or more hot operating modes.

EALs in this category represent threats to the defense in depth design concept that precludes the release of highly radioactive fission products to the environment. This concept relies on multiple physical barriers any one of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment. The primary fission product barriers are:

A. Fuel Clad (FC): The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.
B. Reactor Coolant System (RCS): The RCS Barrier 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. Containment Barrier thresholds are used as criteria for escalation of the ECL from ALERT to a SITE AREA EMERGENCY or a GENERAL EMERGENCY.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 2). "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Alert:
Any loss or any potential loss of either Fuel Clad or RCS

Site Area Emergency:
Loss or potential loss of any two barriers

General Emergency:
Loss of any two barriers and loss or potential loss of third barrier

The logic used for emergency classification based on fission product barrier monitoring should reflect the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier.
- UNUSUAL EVENT ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.
- For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to

ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a SITE AREA EMERGENCY classification while a dose assessment may indicate that an EAL for GENERAL EMERGENCY IC RG1 has been exceeded.

- The fission product barrier thresholds specified within a scheme reflect plant-specific CNP design and operating characteristics.
- As used in this category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside the primary containment, an interfacing system, or outside of the primary containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered to be RCS leakage.
- At the SITE AREA EMERGENCY level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a GENERAL EMERGENCY declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the SEC would have more assurance that there was no immediate need to escalate to a GENERAL EMERGENCY.

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **Any** loss or **any** potential loss of either Fuel Clad or RCS

EAL:

FA1.1	**Alert**
Any loss or **any** potential loss of **EITHER** Fuel Clad **OR** RCS (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the ALERT classification level, Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a SITE AREA EMERGENCY under EAL FS1.1.

CNP 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

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the SITE AREA EMERGENCY classification level, each barrier is weighted equally. A SITE AREA EMERGENCY is therefore appropriate for any combination of the following conditions:

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

At the SITE AREA EMERGENCY classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a GENERAL EMERGENCY is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a GENERAL EMERGENCY classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, the SEC would have greater assurance that escalation to a GENERAL EMERGENCY is less imminent.

CNP 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 third barrier

EAL:

FG1.1	**General Emergency**
Loss of **any** two barriers	
AND	
Loss or potential loss of third barrier (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (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

CNP Basis Reference(s):

1. NEI 99-01 FG1

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. RCS or SG Tube Leakage

B. Inadequate Heat removal

C. CNMT Radiation / RCS Activity

D. CNMT Integrity or Bypass

E. SEC Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word "None" is entered in the cell.

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A would be assigned "FC Loss A.1," the third Containment barrier Potential Loss in Category C would be assigned "CNMT P-Loss C.3," etc.

If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of Table F-1, locates the likely category and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine if a threshold has been exceeded. If a threshold has not been exceeded, the EAL-user proceeds to the next likely category and continues review of the thresholds in the new category

If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded, only that one barrier is lost or potentially lost. The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if containment radiation is sufficiently high, a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Containment barrier can occur. Barrier

Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, and FA1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad barrier threshold bases appear first, followed by the RCS barrier and finally the Containment barrier threshold bases. In each barrier, the bases are given according category Loss followed by category Potential Loss beginning with Category A, then B,..., E.

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad (FC) Barrier		Reactor Coolant System (RCS) Barrier		Containment (CNMT) Barrier	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A RCS or SG Tube Leakage	None	None	1. An automatic or manual ECCS (SI) actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	1. Operation of a standby charging pump is required by **EITHER:** • UNISOLABLE RCS leakage • SG tube leakage 2. CSFST Integrity-RED Path (F-0.4) conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B Inadequate Heat Removal	1. CSFST Core Cooling-RED Path (F-0.2) conditions met	1. CSFST Core Cooling-ORANGE Path (F-0.2) conditions met 2. CSFST Heat Sink-RED Path (F-0.3)conditions met **AND** Heat sink is required	None	1. CSFST Heat Sink-RED Path (F-0.3) conditions met **AND** Heat sink is required	None	1. CSFST Core Cooling-RED Path (F-0.2) conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C CNMT Radiation / RCS Activity	1. Containment radiation > Table F-2 column "FC Loss" 2. Dose equivalent I-131 coolant activity > 300 μCi/cc	None	1. Containment radiation > Table F-2 column "RCS Loss"	None	None	1. Containment radiation > Table F-2 column "CNMT Potential Loss"
D CNMT Integrity or Bypass	None	None	None	None	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on SEC judgment • UNISOLABLE pathway from containment to the environment exists 2. Indications of RCS leakage outside of Containment	1. CSFST Containment-RED Path (F-0.5) conditions met 2. Containment hydrogen concentration ≥ 4% 3. Containment pressure > 2.8 psig with < one full train of depressurization equipment operating per design for ≥ 15 min. (Note 1, 9)
E SEC Judgment	1. Any condition in the opinion of the SEC that indicates loss of the Fuel Clad barrier	1. Any condition in the opinion of the SEC that indicates potential loss of the Fuel Clad barrier	1. Any condition in the opinion of the SEC that indicates loss of the RCS barrier	1. Any condition in the opinion of the SEC that indicates potential loss of the RCS barrier	1. Any condition in the opinion of the SEC that indicates loss of the Containment barrier	1. Any condition in the opinion of the SEC that indicates potential loss of the Containment barrier

Barrier: Fuel Clad

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

1. CSFST Core Cooling-RED Path (F-0.2) conditions met

Definition(s):

None

Basis:

Indication of continuing severe core cooling degradation is manifested by CSFST Core Cooling RED PATH conditions being met. Specifically, Core Cooling RED PATH conditions exist if either the five highest core exit TCs are reading greater than or equal to 1200°F or core exit TCs are reading greater than or equal to 757°F with RCS subcooling less than or equal 40°F and RVLIS level less than or equal to that specified based on the number of RCPs running (ref. 1).

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path indicates significant core exit superheating and core uncovery. The CSFSTs are normally monitored using the SPDS display on the Plant Process Computer (ref. 1, 2).

This reading indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

CNP Basis Reference(s):

1. 1(2)-OHP04023-F-0.2 Critical Safety Function Status Trees – Core Cooling
2. 1(2)-OHP-4023-FR-C.1 Response to Inadequate Core Cooling
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Core Cooling-ORANGE Path (F-0.2) conditions met

Definition(s):

None

Basis:

Indication of continuing significant core cooling degradation is manifested by CSFST Core Cooling ORANGE PATH conditions being met. Specifically, Core Cooling ORANGE PATH conditions exist if either the five highest core exit TCs are reading greater than or equal to 757°F with RCS subcooling less than or equal 40°F or RVLIS level less than or equal to that specified based on the number of RCPs running (ref. 1).

Critical Safety Function Status Tree (CSFST) Core Cooling-ORANGE path indicates subcooling has been lost and that some fuel clad damage may potentially occur. The CSFSTs are normally monitored using the SPDS display on the Plant Computer (ref. 1, 2).

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

CNP Basis Reference(s):

1. 1(2)-OHP04023-F-0.2 Critical Safety Function Status Trees – Core Cooling
2. 1(2)-OHP-4023-FR-C.2 Response to Degraded Core Cooling
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. CSFST Heat Sink-RED Path (F-0.3) conditions met **AND** Heat sink is required

Definition(s):

None

Basis:

In combination with RCS Potential Loss B.1, meeting this threshold results in a SITE AREA EMERGENCY.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED path indicates the ultimate heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

Heat Sink RED PATH conditions exist if narrow range level in all SGs is less than or equal to 13% and total feedwater flow to all SGs is less than or equal to 240,000 lbm/hr (ref. 1).

The CSFSTs are normally monitored using the SPDS display on the Plant Computer (ref. 2).

The phrase "and heat sink required" precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. For example, FR-H.1 is entered from CSFST Heat Sink-Red. Step 1 tells the operator to determine if heat sink is required by checking that RCS pressure is greater than any non-faulted SG pressure and RCS temperature is greater than 350°F. If these conditions exist, Heat Sink is required. Otherwise, the operator is to either return to the procedure and step in effect and place RHR in service for heat removal. For large LOCA events inside the Containment, the SGs are moot because heat removal through the containment heat removal systems takes place. Therefore, Heat Sink Red should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an ALERT classification. (ref. 2).

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the Fuel Clad Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

CNP Basis Reference(s):

1. 1(2)-OHP04023-F-0.3 Critical Safety Function Status Trees – Heat Sink
2. 1(2)-OHP-4023-FR-H.1 Response to Loss of Secondary Heat Sink
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.B

Barrier: Fuel Clad

Category: C. CNMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment radiation > Table F-2 column "FC Loss"

Table F-2 Containment Radiation – R/hr - VRA-1310 (2310) / 1410 (2410)			
Monitor	**FC Loss**	**RCS Loss**	**CNMT Potential Loss**
VRA-1310 (2310)	1,000	200	9,100
VRA-1410 (2410)	700	140	6,300

Definition(s):

None

Basis:

Containment radiation monitor readings greater than Table F-2 column "FC Loss" (ref. 1) indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading is derived assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/cc dose equivalent I-131 into the containment atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage (2 - 3% clad failure depending on core inventory and RCS volume). This value is higher than that specified for RCS barrier Loss C.1 (ref. 1, 2).

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors CHRM-VRA-1310/1410 (2310/2410).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 3% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold C.1 since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the ECL to a SITE AREA EMERGENCY.

CNP Basis Reference(s):

1. EP-CALC-CNP-1602, Containment Radiation EAL Threshold Values
2. EVAL-RD-99-11, Evaluation of Radiation Monitoring System Setpoints, Rev 0
3. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.A

Barrier: Fuel Clad

Category: C. CNMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. Dose equivalent I-131 coolant activity > 300 μCi/gm

Definition(s):

None

Basis:

This threshold indicates that RCS radioactivity concentration is greater than 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 3% fuel clad damage (ref. 1). Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications.0

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

CNP Basis Reference(s):

1. EP-EALCALC-CNP-1602, Containment Radiation EAL Threshold Values
2. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.B

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: C. CNMT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CNMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: D. CNMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. SEC Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the SEC that indicates loss of the Fuel Clad barrier

Definition(s):

None

Basis:

The SEC judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The SEC should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that are to be used by the SEC in determining whether the Fuel Clad barrier is lost

CNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Fuel Clad Loss 6.A

Barrier: Fuel Clad

Category: E. SEC Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the SEC that indicates potential loss of the Fuel Clad barrier

Basis:

The SEC judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The SEC should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that are to be used by the SEC in determining whether the Fuel Clad barrier is potentially lost. The SEC should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

CNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Potential Fuel Clad Loss 6.A

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. An automatic or manual ECCS (SI) actuation required by **EITHER:**
   - UNISOLABLE RCS leakage
   - SG tube RUPTURE

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

RUPTURE - The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

Basis:

ECCS (SI) actuation is caused by (ref. 1, 2):

- Pressurizer low pressure
- Steamline low pressure
- Lower Containment high pressure
- Steamline ΔP

This threshold is based on an UNISOLABLE RCS leak of sufficient size to require an automatic or manual actuation of the Emergency Core Cooling System (ECCS). This condition clearly represents a loss of the RCS Barrier.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED. If a RUPTURED steam generator is also FAULTED outside of containment, the declaration escalates to a SITE AREA EMERGENCY since the Containment Barrier Loss threshold A.1 will also be met.

CNP Basis Reference(s):

1. 1(2)-OHP-4023-E-0 Reactor Trip or Safety Injection
2. 1(2)-OHP-4023-E-3 Steam Generator Tube Rupture
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Loss 1.A

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

1. Operation of a standby charging pump is required by **EITHER:**
    - UNISOLABLE RCS leakage
    - SG tube leakage

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

This threshold is based on the inability to maintain liquid inventory within the RCS by normal operation of the Chemical and Volume Control System (CVCS). The CVCS includes three charging pumps: one positive displacement pump with a flow capacity of 150 gpm, and two centrifugal charging pumps each with a flow capacity of 150 gpm (ref. 1). A second charging pump being required is indicative of a substantial RCS leak.

This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ECCS (SI) actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs that a standby charging (makeup) pump be placed in service to restore and maintain pressurizer level.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

If a leaking steam generator is also FAULTED outside of containment, the declaration escalates to a SITE AREA EMERGENCY since the Containment Barrier Loss threshold 1.A will also be met.

CNP Basis Reference(s):

1. UFSAR Table 9.2-2 Chemical and Volume Control System Design Parameters
2. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.A

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

2. CSFST Integrity-RED Path (F-0.4) conditions met

Definition(s):

None

Basis:

The "Potential Loss" threshold is defined by the CSFST Reactor Coolant Integrity - RED path. CSFST RCS Integrity - Red Path plant conditions and associated PTS Limit Curve A indicates an extreme challenge to the safety function when plant parameters are to the left of the limit curve following excessive RCS cooldown under pressure (ref. 1, 2).

This condition indicates an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized).

CNP Basis Reference(s):

1. 1(2)-OHP-4023-F-0.4 Critical Safety Function Status Trees Figure F-0.4-1 Integrity Operational Limits
2. 1(2)-OHP-4023-FR-P.1 Response to Imminent Pressurized Thermal Shock Condition
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.B

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Heat Sink-RED Path (F-0.3) conditions met

   AND

   Heat sink is required

Definition(s):

None

Basis:

In combination with FC Potential Loss B.2, meeting this threshold results in a SITE AREA EMERGENCY.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED path indicates the ultimate heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

Heat Sink RED PATH conditions exist if narrow range level in all SGs is less than or equal to 13% and total feedwater flow to all SGs is less than or equal to 240,000 lbm/hr (ref. 1).

The CSFSTs are normally monitored using the SPDS display on the Plant Computer (ref. 2).

The phrase "and heat sink required" precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. For example, FR-H.1 is entered from CSFST Heat Sink-Red. Step 1 tells the operator to determine if heat sink is required by checking that RCS pressure is greater than any non-faulted SG pressure and RCS temperature is greater than 350ºF. If these conditions exist, Heat Sink is required. Otherwise, the operator is to either return to the procedure and step in effect and place RHR in service for heat removal. For large LOCA events inside the Containment, the SGs are moot because heat removal through the containment heat removal systems takes place. Therefore, Heat Sink Red should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an ALERT classification. (ref. 2).

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the RCS Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

Meeting this threshold results in a SITE AREA EMERGENCY because this threshold is identical to Fuel Clad Barrier Potential Loss threshold B.2; both will be met. This condition warrants a SITE AREA EMERGENCY declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase RCS pressure to the point where mass will be lost from the system.

CNP Basis Reference(s):

1. 1(2)-OHP04023-F-0.3 Critical Safety Function Status Trees – Heat Sink
2. 1(2)-OHP-4023-FR-H.1 Response to Loss of Secondary Heat Sink
3. NEI 99-01 Inadequate Heat Removal RCS Loss 2.B

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: C. CNMT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment radiation > Table F-2 column "RCS Loss"

Table F-2 Containment Radiation – R/hr - VRA-1310 (2310) / 1410 (2410)			
Monitor	**FC Loss**	**RCS Loss**	**CNMT Potential Loss**
VRA-1310 (2310)	1,000	200	9,100
VRA-1410 (2410)	700	140	6,300

Definition(s):

N/A

Basis:

Containment radiation monitor readings greater than Table F-2 column "RCS Loss" (ref. 1, 2) indicate the release of reactor coolant to the containment. The readings assume 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 primary coolant.

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors CHRM-VRA-1310/1410 (2310/2410).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold C.1 since it indicates a loss of the RCS Barrier only.

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

CNP Basis Reference(s):

1. EP-CALC-CNP-1602, Containment Radiation EAL Threshold Values
2. EVAL-RD-99-11, Evaluation of Radiation Monitoring System Setpoints, Rev 0
3. NEI 99-01 CMT Radiation / RCS Activity RCS Loss 3.A

Barrier: Reactor Coolant System

Category: B. CNMT Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CNMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CNMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. SEC Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the SEC that indicates loss of the RCS barrier

Definition(s):

None

Basis:

The SEC judgment threshold addresses any other factors relevant to determining if the RCS barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The SEC should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the SEC in determining whether the RCS Barrier is lost.

CNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Loss 6.A

Barrier: Reactor Coolant System

Category: E. SEC Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the SEC that indicates potential loss of the RCS barrier

Definition(s):

None

Basis:

The SEC judgment threshold addresses any other factors relevant to determining if the RCS barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The SEC should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the SEC in determining whether the RCS Barrier is potentially lost. The SEC should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

CNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Potential Loss 6.A

Barrier: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. A leaking or RUPTURED SG is FAULTED outside of containment

Definition(s):

FAULTED - The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.

RUPTURED - The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

Basis:

This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for RCS Barrier Potential Loss A.1 and Loss A.1, respectively. This condition represents a bypass of the containment barrier.

FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably (part of the FAULTED definition) and the FAULTED steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition, the steam generator is still considered FAULTED for emergency classification purposes.

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU4 for the fuel clad barrier (i.e., RCS activity values) and IC SU5 for the RCS barrier (i.e., RCS leak rate values).

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, glad seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

The ECLs resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

	Affected SG is FAULTED Outside of Containment?	
P-to-S Leak Rate	**Yes**	**No**
Less than or equal to 25 gpm	No classification	No classification
Greater than 25 gpm	UNUSUAL EVENT per SU5.1	UNUSUAL EVENT per SU5.1
Requires operation of a standby charging (makeup) pump (*RCS Barrier Potential Loss*)	SITE AREA EMERGENCY per FS1.1	ALERT per FA1.1
Requires an automatic or manual ECCS (SI) actuation (*RCS Barrier Loss*)	SITE AREA EMERGENCY per FS1.1	ALERT per FA1.1

There is no Potential Loss threshold associated with RCS or SG Tube Leakage.

CNP Basis Reference(s):

1. NEI 99-01 RCS or SG Tube Leakage Containment Loss 1.A

Barrier: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: B. Inadequate heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: B. Inadequate heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Core Cooling-RED Path (F-0.2) conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Definition(s):

None

Basis:

Indication of continuing severe core cooling degradation is manifested by CSFST Core Cooling RED PATH conditions being met. Specifically, Core Cooling RED PATH conditions exist if either the five highest core exit TCs are reading greater than or equal to 1200°F or core exit TCs are reading greater than or equal to 757°F with RCS subcooling less than or equal 40°F and RVLIS level less than or equal to that specified based on the number of RCPs running (ref. 1).

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path indicates significant core exit superheating and core uncovery. The CSFSTs are normally monitored using the SPDS display on the Plant Computer (ref. 1).

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 (ref. 1, 2, 3).

A direct correlation to status trees can be made if the effectiveness of the restoration procedures is also evaluated. If core exit thermocouple (TC) readings are greater than 1,200°F (ref. 1), Fuel Clad barrier is also lost.

This threshold addresses any other factors that may be used by the SEC in determining whether the RCS Barrier is potentially lost. The SEC should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

CNP Basis Reference(s):

1. 1(2)-OHP04023-F-0.2 Critical Safety Function Status Trees – Core Cooling
2. 1(2)-OHP-4023-FR-C.1 Response to Inadequate Core Cooling
3. 1(2)-OHP04023-FR-C.2 Response to Degraded Core Cooling
4. NEI 99-01 Inadequate Heat Removal Containment Potential Loss 2.A

Barrier: Containment

Category: C. CNMT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: C. CNMT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

1. Containment radiation > Table F-2 column "CNMT Potential Loss"

Table F-2 Containment Radiation – R/hr - VRA-1310 (2310) / 1410 (2410)			
Monitor	**FC Loss**	**RCS Loss**	**CNMT Potential Loss**
VRA-1310 (2310)	1,000	200	9,100
VRA-1410 (2410)	700	140	6,300

Definition(s):

None

Basis:

Containment radiation monitor readings greater than Table F-2 column "CNMT Potential Loss" (ref. 1, 2) indicate significant fuel damage (20% clad damage) well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier.

The readings are higher than that specified for Fuel Clad barrier Loss C.1 and RCS barrier Loss C.1. 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.

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors CHRM-VRA-1310/1410 (2310/2410).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the ECL to a GENERAL EMERGENCY.

CNP Basis Reference(s):

1. EP-CALC-CNP-1602, Containment Radiation EAL Threshold Values
2. EVAL-RD-99-11, Evaluation of Radiation Monitoring System Setpoints, Rev 0
3. NEI 99-01 CMT Radiation / RCS Activity Containment Potential Loss 3.A

Barrier: Containment

Category: D. CNMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. Containment isolation is required

   AND EITHER:

   - Containment integrity has been lost based on SEC judgment
   - UNISOLABLE pathway from containment to the environment exists

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold A.1.

These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both bulleted thresholds.

First Threshold – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the SEC will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure 1. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

Another example would be a loss or potential loss of the RCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

Second Threshold – Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment atmosphere to the environment. As used here, the term "environment" includes the atmosphere of a room or area, outside the containment, that may, in turn, communicate with the outside-the-plant atmosphere (e.g., through discharge of a ventilation system or atmospheric leakage). Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.

Refer to the top piping run of Figure 1. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Leakage between two interfacing liquid systems, by itself, does not meet this threshold.

Refer to the bottom piping run of Figure 1. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump developed a leak that allowed steam/water to enter the Auxiliary Building, then second threshold would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause the first threshold to be met as well.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable containment leakage through various penetrations or system components. Minor releases may also occur if a containment isolation valve(s) fails to close but the containment atmosphere escapes to an enclosed system. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

CNP Basis Reference(s):

1. NEI 99-01 CMT Integrity or Bypass Containment Loss 4.A

Barrier: Containment

Category: D. CNMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

2. Indications of RCS leakage outside of containment

Definition(s):

None

Basis:

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold A.1.

To ensure proper escalation of the emergency classification, the RCS leakage outside of containment must be related to the mass loss that is causing the RCS Loss and/or Potential Loss threshold A.1 to be met.

ECA-1.2 LOCA Outside Containment (ref. 1) provides instructions to identify and isolate a LOCA outside of the containment. Potential RCS leak pathways outside containment include (ref. 1, 2):

- Residual Heat Removal
- Safety Injection
- Chemical & Volume Control
- RCP seals

Containment sump, temperature, pressure and/or radiation levels will increase if reactor coolant mass is leaking into the containment. If these parameters have not increased, then the reactor coolant mass may be leaking outside of containment (i.e., a containment bypass sequence). Increases in sump, temperature, pressure, flow and/or radiation level readings outside of the containment may indicate that the RCS mass is being lost outside of containment.

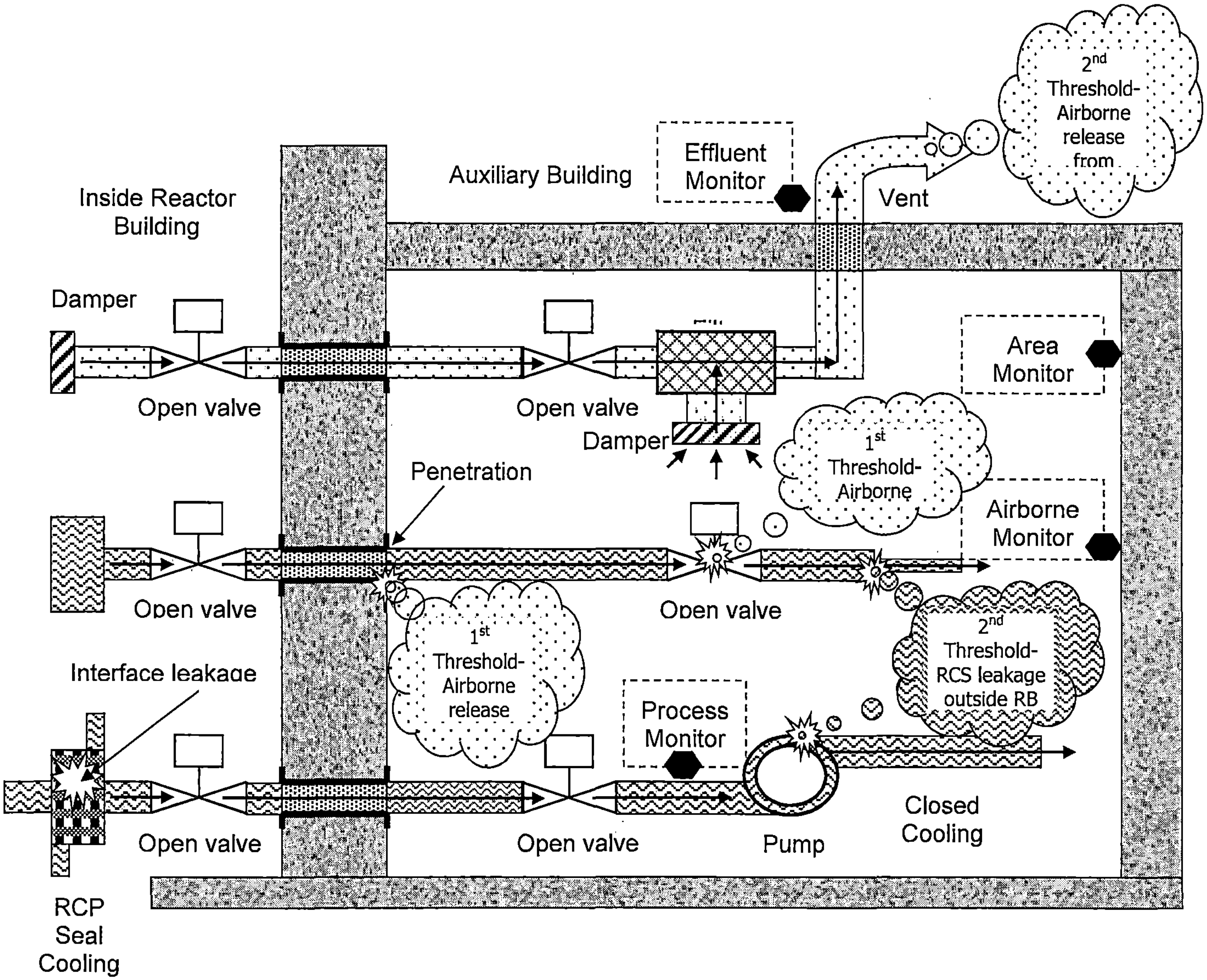
Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of RCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside of containment may not increase significantly; however, other unexpected changes in sump levels, area temperatures or pressures, flow rates, etc. should be sufficient to determine if RCS mass is being lost outside of the containment.

Refer to the middle piping run of Figure 1. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause threshold D.1 to be met as well.

CNP Basis Reference(s):

1. 1(2)-OHP-4023-ECA-1.2 LOCA Outside Containment
2. 1(2)-OHP-4023-E-1 Loss of Reactor or Secondary Coolant
3. NEI 99-01 CMT Integrity or Bypass Containment Loss

Figure 1: Containment Integrity or Bypass Examples



Barrier: Containment

Category: D. CNMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

1. CSFST Containment-RED Path (F-0.5) conditions met

Definition(s):

None

Basis:

Critical Safety Function Status Tree (CSFST) Containment-RED path is entered if containment pressure is greater than or equal to 12 psig and represents an extreme challenge to safety function. The CSFSTs are normally monitored using the SPDS display on the Plant Computer (ref. 1, 2).

12 psig is the containment design pressure (ref. 3) and is the pressure used to define CSFST Containment Red Path conditions.

If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the RCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a SITE AREA EMERGENCY and GENERAL EMERGENCY since there is now a potential to lose the third barrier.

CNP Basis Reference(s):

1. 1(2)-OHP-4032-F00.5 Critical Safety Function Status Trees Containment
2. 1(2)-OHP-4023-FR-Z.1 Response to High Containment Pressure
3. UFSAR Section 5.2.2.2 Design Load Criteria
4. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.A

Barrier: Containment

Category: D. CNMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

2. Containment hydrogen concentration ≥ 4%

Definition(s):

None

Basis:

Following a design basis accident, hydrogen gas may be generated inside the containment by reactions such as zirconium metal with water, corrosion of materials of construction and radiolysis of aqueous solution in the core and sump. The lower limit of combustion of hydrogen in air is approximately 4%.

CNP is equipped with a Post-Accident Hydrogen Monitoring System (PACHMS) which serves to measure combustible gas concentrations in the containment. The PACHMS is comprised of two sampling-analyzing-control trains (ref. 1).

To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers must have occurred. With the Potential Loss of the containment barrier, the threshold hydrogen concentration, therefore, will likely warrant declaration of a GENERAL EMERGENCY.

Two Containment hydrogen monitors with duel ranges of 0% to 10% and 0% to 30% provide indication locally and in the Control Room (ref. 1, 2).

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a potential loss of the Containment Barrier.

CNP Basis Reference(s):

1. UFSAR Section 7.8.2 Post-Accident Hydrogen Monitoring
2. 12-THP-6020-PAS-003, Post Accident Containment Hydrogen Monitoring System Operation
3. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.B

Barrier: Containment

Category: D. CNMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

3. Containment pressure > 2.8 psig with < one full train of containment depressurization equipment operating per design for ≥ 15 min. (Notes 1, 9)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 9: One Containment Spray System train and one Containment Air Recirculation Fan comprise one full train of depressurization equipment.

Definition(s):

None

Basis:

Containment pressure control is achieved through the Containment Spray System and the Containment Air Recirculation/Hydrogen Skimmer System. Failure of either of these systems may allow steam to build up within containment, and, unabated, this steam buildup may cause the internal containment pressure buildup to exceed the design pressure of 12 psig. Studies have shown that the containment can withstand pressures well above this value.

Both the recirculation fans and the containment spray pumps are actuated automatically (delayed) following receipt of a HI or HI HI containment pressure signal, respectively. Since the HI HI containment pressure setpoint is less than or equal to 2.8 PSI, then greater than 2.8 PSI would be the containment pressure greater than the setpoint at which the equipment was supposed to have actuated. If these systems should fail to start automatically per design, a successful manual start within 15 minutes would preclude exceeding this Containment Potential Loss threshold. (ref. 1, 2, 3).

This threshold describes a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems (e.g., containment sprays, containment recirculation fans, etc., but not including containment venting strategies) are either lost or performing in a degraded manner.

CNP Basis Reference(s):

1. UFSAR Section 5.5.3 System Description
2. UFSAR Section 6.3 Containment Spray Systems
3. EC-0000052930 Unit 1 Return to Normal Operating Pressure and Temperature (NOP/NOT)
4. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.C

Barrier: Containment

Category: E. SEC Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the SEC that indicates loss of the Containment barrier

Definition(s):

None

Basis:

The SEC judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The SEC should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the SEC in determining whether the Containment Barrier is lost.

CNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

Barrier: Containment

Category: E. SEC Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the SEC that indicates potential loss of the Containment barrier

Definition(s):

None

Basis:

The SEC judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The SEC should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the SEC in determining whether the Containment Barrier is lost.

CNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Potential Loss 6.A

Background

NEI 99-01 Revision 6 ICs AA3 and HA5 prescribe declaration of an ALERT based on impeded access to rooms or areas (due to either area radiation levels or hazardous gas concentrations) where equipment necessary for normal plant operations, cooldown or shutdown is located. These areas are intended to be plant operating mode dependent. Specifically the Developers Notes For AA3 and HA5 states:

The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.

The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).

Further, as specified in IC HA5:

The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.

CNP Table R-2 and H-2 Bases

A review of station operating procedures identified the following mode dependent in-plant actions and associated rooms or areas that are required for normal plant operation, cooldown or shutdown:

In-Plant Actions	Safe Shutdown Area	Modes
Condensate, Feedwater & Main / Feed Pump Turbine Operating Procedures	Turbine Building All Levels	1, 2, 3
Reactor Coolant Sampling	Auxiliary 587' & Aux 609'	1, 2, 3, 4, 5
Steam Generator Blowdown System Operation	Auxiliary 587', 591', & 633'	1, 2, 3, 4
Operation of Screen Wash and Traveling Screens	Screenhouse	1, 2, 3, 4, 5
Auxiliary Feedwater Pump Operations	Turbine Building 591'	1, 2, 3, 4, 5
Operation of the Residual Heat Removal System	Auxiliary 573' & Aux 609'	4, 5
ECCS Breaker Alignments	Auxiliary 587', 609' & 633'	5

Table R-2 & H-2 Results

Table R-2 & H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Auxiliary Building 573'	4, 5
Auxiliary Building 587' (including D/G room)	1, 2, 3, 4, 5
Auxiliary Building 591'	1, 2, 3, 4
Auxiliary Building 609' (including 4kV room)	1, 2, 3, 4, 5
Auxiliary Building 633'	1, 2, 3, 4
Turbine Building (**All** Levels)	1, 2, 3
Turbine Building 591'	4, 5
Screenhouse	1, 2, 3, 4, 5

Plant Operating Procedures Reviewed

Mode	Procedure	Operation
1	OHP-4021-011-001	At-Power Operation Including Load Swings
1-3	OHP-4021-001-003	Power Reduction
1-5	THP-6020-CHM-121	Reactor Coolant Sampling
1-4	OHP-4021-025-001	Steam Generator Blowdown
3-5	OHP-4021-001-004	Plant Cooldown from Hot Standby to Cold Shutdown
4-5	OHP-4021-017-002	Placing in Service the Residual Heat Removal System
1-5	OHP-4021-056-001	Auxiliary Feed Pump Operation
1-5	OHP-4021-057-005	Operation of Screen Wash and Traveling Screens
1-5	OHP-4021-019-001	Operation of the Essential Service Water System
1-5	OHP-4021-018-002	Placing in Service and Operating the Spent Fuel Pit Cooling and Cleanup System

Enclosure 5 to AEP-NRC-2017-02

DONALD C. COOK NUCLEAR PLANT EMERGENCY PLAN
NEI 99-01 REVISION 6 EAL COMPARISON MATRIX

AEP: D.C. Cook
NEI 99-01 Revision 6
EAL Comparison Matrix

Revision 0

Table of Contents

Introduction

This document provides a line-by-line comparison of the Initiating Conditions (ICs), Mode Applicability and Emergency Action Levels (EALs) in NEI 99-01 Rev. 6 Final, Development of Emergency Action Levels for Non-Passive Reactors, ADAMS Accession Number ML12328A805, and the D. C. Cook Nuclear Plant (CNP) ICs, Mode Applicability and EALs. This document provides a means of assessing CNP differences and deviations from the NRC endorsed guidance given in NEI 99-01. Discussion of CNP EAL bases and lists of source document references are given in the EAL Technical Bases Document. It is, therefore, advisable to reference the EAL Technical Bases Document for background information while using this document.

Comparison Matrix Format

The ICs and EALs discussed in this document are grouped according to NEI 99-01 Recognition Categories. Within each Recognition Category, the ICs and EALs are listed in tabular format according to the order in which they are given in NEI 99-01. Generally, each row of the comparison matrix provides the following information:

- NEI EAL/IC identifier
- NEI EAL/IC wording
- CNP EAL/IC identifier
- CNP EAL/IC wording
- Description of any differences or deviations

EAL Wording

In Section 4.1, NEI recommends the following: "The guidance in NEI 99-01 is not intended to be applied to plants "as-is"; however, developers should attempt to keep their site-specific schemes as close to the generic guidance as possible. The goal is to meet the intent of the generic Initiating Conditions (ICs) and Emergency Action Levels (EALs) within the context of site-specific characteristics – locale, plant design, operating features, terminology, etc. Meeting this goal will result in a shorter and less cumbersome NRC review and approval process, closer alignment with the schemes of other nuclear power plant sites and better positioning to adopt future industry-wide scheme enhancements"

To assist the Site Emergency Coordinator (SEC), the CNP EALs have been written in a clear and concise style (to the extent that the differences from the NEI EAL wording could be reasonably documented and justified

EAL Emphasis Techniques

Due to the width of the table columns and table formatting constraints in this document, line breaks and indentation may differ slightly from the appearance of comparable wording in the source documents. NEI 99-01 is the source document for the NEI EALs; the CNP EAL Technical Bases Document for the CNP EALs.

Development of the CNP IC/EAL wording has attempted to minimize inconsistencies and apply sound human factors principles. As a result, differences occur between NEI and CNP ICs/EALs for these reasons alone. When such difference may infer a technical difference in the associated NEI IC/EAL, the difference is identified and a justification provided.

The print and paragraph formatting conventions summarized below guide presentation of the CNP EALs in accordance with the EAL writing criteria. Space restrictions in the EAL table of this document sometimes override this criteria in cases when following the criteria would introduce undesirable complications in the EAL layout.

- Upper case-bold print is used for the logic terms **AND**, **OR** and **EITHER**.
- Bold font is used for certain logic terms, negative terms (**not**, **cannot**, etc.), **any**, **all**.
- Upper case print is reserved for defined terms, acronyms, system abbreviations, logic terms (and, or, etc. when not used as a conjunction), annunciator window engravings.
- Three or more items in a list are normally introduced with "**Any** of the following..." or "**All** of the following..." Items of the list begin with bullets when a priority or sequence is not inferred.
- The use of **AND/OR** logic within the same EAL has been avoided when possible. When such logic cannot be avoided, indentation and separation of subordinate contingent phrases is employed.

Global Differences

The differences listed below generally apply throughout the set of EALs and are not repeated in the Justification sections of this document. The global differences do not decrease the effectiveness of the intent of NEI 99-01.

1. The NEI phrase "Notification of Unusual Event" has been changed to "Unusual Event" or abbreviated "UE" to reduce EAL-user reading burden.
2. NEI 99-01 IC Example EALs are implemented in separate plant EALs to improve clarity and readability. For example, NEI lists all IC HU3 Example EALs under one IC. The corresponding CNP EALs appear as unique EALs (e.g., HU3.1 through HU3.4).
3. Mode applicability identifiers (numbers/letter) modify the NEI 99-01 mode applicability names as follows: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown, 5 - Cold Shutdown, 6 - Refueling, D – Defueled. NEI 99-01defines Defueled as follows: "Reactor Vessel contains no irradiated fuel (full core off-load during refueling or extended outage)."
4. NEI 99-01 uses the terms greater than, less than, greater than or equal to, etc. in the wording of some example EALs. For consistency and reduce EAL-user reading burden, CNP has adopted use of boolean symbols in place of the NEI 99-01 text modifiers within the EAL wording.
5. "min." is the standard abbreviation for "minutes" and is used to reduce EAL user reading burden.
6. The term "Emergency Director" has been replaced by "Site Emergency Coordinator (SEC)" consistent with site-specific nomenclature.
7. Wherever the generic bracketed PWR term "reactor vessel/RCS" is provided, CNP uses the term "RCS" as the site-specific nomenclature.
8. IC/EAL identification:
   - NEI Recognition Category A "Abnormal Radiation Levels/ Radiological Effluents" has been changed to Category R "Abnormal Rad Levels / Rad Effluents." The designator "R" is more intuitively associated with radiation (rad) or radiological events. NEI IC designators beginning with "A" have likewise been changed to "R."
   - The CNP IC/EAL scheme includes the following features:
     a. Division of the NEI EAL set into three groups:
        - EALs applicable under <u>all</u> plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
        - EALs applicable only under <u>hot</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Hot Standby, Startup or Power Operation mode.
        - EALs applicable only under <u>cold</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling or Defueled mode.

        The purpose of the groups is to avoid review of hot condition EALs when the plant is in a cold condition and avoid review of cold condition EALs when the plant is in a hot condition. This approach significantly minimizes the total number of EALs that must be reviewed by the EAL-user for a given plant condition and, thereby, speeds identification of the EAL that applies to the emergency.
     b. Within each of the above three groups, assignment of EALs to categories/subcategories – Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. Subcategories are used as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The CNP EAL categories/subcategories and their relationship to NEI Recognition Categories are listed in Table 1.
     c. Unique identification of each EAL – Four characters comprise the EAL identifier as illustrated in Figure 1.

Figure 1 – EAL Identifier

EAL Identifier

XXX.X

Category (R, H, E, S, F, C)

Emergency classification (G, S, A, U)

Sequential number within subcategory/classification

Subcategory number (1 if no subcategory)

The first character is a letter associated with the category in which the EAL is located. The second character is a letter associated with the emergency classification level (G for General Emergency, S for Site Area Emergency, A for Alert, and U for Notification of Unusual Event). The third character is a number associated with one or more subcategories within a given category. Subcategories are sequentially numbered beginning with the number "1". If a category does not have a subcategory, this character is assigned the number "1". The fourth character is a number preceded by a period for each EAL within a subcategory. EALs are sequentially numbered within the emergency classification level of a subcategory beginning with the number "1".

The EAL identifier is designed to fulfill the following objectives:

- Uniqueness – The EAL identifier ensures that there can be no confusion over which EAL is driving the need for emergency classification.
- Speed in locating the EAL of concern – When the EALs are displayed in a matrix format, knowledge of the EAL identifier alone can lead the EAL-user to the location of the EAL within the classification matrix. The identifier conveys the category, subcategory and classification level. This assists ERO responders (who may not be in the same facility as the ED) to find the EAL of concern in a timely manner without the need for a word description of the classification threshold.
- Possible classification upgrade – The category/subcategory/identifier scheme helps the EAL-user find higher emergency classification EALs that may become active if plant conditions worsen.

Table 2 lists the CNP ICs and EALs that correspond to the NEI ICs/Example EALs when the above EAL/IC organization and identification scheme is implemented.

Differences and Deviations

In accordance NRC Regulatory Issue Summary (RIS) 2003-18 "Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels" Supplements 1 and 2, a difference is an EAL change in which the basis scheme guidance differs in wording but agrees in meaning and intent, such that classification of an event would be the same, whether using the basis scheme guidance or the CNP EAL. A deviation is an EAL change in which the basis scheme guidance differs in wording and is altered in meaning or intent, such that classification of the event could be different between the basis scheme guidance and the CNP proposed EAL.

Administrative changes that do not actually change the textual content are neither differences nor deviations. Likewise, any format change that does not alter the wording of the IC or EAL is considered neither a difference nor a deviation.

The following are examples of differences:

- Choosing the applicable EAL based upon plant type (i.e., BWR vs. PWR).
- Using a numbering scheme other than that provided in NEI 99-01 that does not change the intent of the overall scheme.
- Where the NEI 99-01 guidance specifically provides an option to not include an EAL if equipment for the EAL does not exist at CNP (e.g., automatic real-time dose assessment capability).
- Pulling information from the bases section up to the actual EAL that does not change the intent of the EAL.
- Choosing to state ALL Operating Modes are applicable instead of stating N/A, or listing each mode individually under the Abnormal

Rad Level/Radiological Effluent and Hazard and Other Conditions Affecting Plant Safety sections.

- Using synonymous wording (e.g., greater than or equal to vs. at or above, less than or equal vs. at or below, greater than or less than vs. above or below, etc.)
- Adding CNP equipment/instrument identification and/or noun names to EALs.
- Combining like ICs that are exactly the same but have different operating modes as long as the intent of each IC is maintained and the overall progression of the EAL scheme is not affected.
- Any change to the IC and/or EAL, and/or basis wording, as stated in NEI 99-01, that does not alter the intent of the IC and/or EAL, i.e., the IC and/or EAL continues to:
  - Classify at the correct classification level.
  - Logically integrate with other EALs in the EAL scheme.
  - Ensure that the resulting EAL scheme is complete (i.e., classifies all potential emergency conditions).

The following are examples of deviations:

- Use of altered mode applicability.
- Altering key words or time limits.
- Changing words of physical reference (protected area, safety-related equipment, etc.).
- Eliminating an IC. This includes the removal of an IC from the Fission Product Barrier Degradation category as this impacts the logic of Fission Product Barrier ICs.
- Changing a Fission Product Barrier from a Loss to a Potential Loss or vice-versa.
- Not using NEI 99-01definitions as the intent is for all NEI 99-01 users to have a standard set of defined terms as defined in NEI 99-01. Differences due to plant types are permissible (BWR or PWR). Verbatim compliance to the wording in NEI 99-01 is not necessary as long as the intent of the defined word is maintained. Use of the wording provided in NEI 99-01 is encouraged since the intent is for all users to have a standard set of defined terms as defined in NEI 99-01.
- Any change to the IC and/or EAL, and/or basis wording as stated in NEI 99-01 that does alter the intent of the IC and/or EAL, i.e., the IC and/or EAL:
  - Does not classify at the classification level consistent with NEI 99-01.
  - Is not logically integrated with other EALs in the EAL scheme.
  - Results in an incomplete EAL scheme (i.e., does not classify all potential emergency conditions).

The "Difference/Deviation Justification" columns in the remaining sections of this document identify each difference between the NEI 99-01 IC/EAL wording and the CNP IC/EAL wording. An explanation that justifies the reason for each difference is then provided. If the difference is determined to be a deviation, a statement is made to that affect and explanation is given that states why classification may be different from the NEI 99-01 IC/EAL and the reason for its acceptability. In all cases, however, the differences and deviations do not decrease the effectiveness of the intent of NEI 99-01. CNP has identified two deviations from the NEI 99-01 guidance as represented in Table 3.

Table 1 – CNP EAL Categories/Subcategories

CNP EALs		NEI Recognition Category
Category	**Subcategory**	
Group: Any Operating Mode:		
R – Abnormal Rad Levels/Rad Effluent	1 – Radiological Effluent 2 – Irradiated Fuel Event 3 – Area Radiation Levels	Abnormal **R**ad Levels/Radiological Effluent ICs/EALs
H – Hazards and Other Conditions Affecting Plant Safety	1 – Security 2 – Seismic Event 3 – Natural or Technological Hazard 4 – Fire 5 – Control Room Evacuation 6 – SEC Judgment	**H**azards and Other Conditions Affecting Plant Safety ICs/EALs
E - ISFSI	1 – Confinement Boundary	ISFSI ICs/EALs
Group: Hot Conditions:		
S – System Malfunction	1 – Loss of Emergency AC Power 2 – Loss of Vital DC Power 3 – Loss of Control Room Indications 4 – RCS Activity 5 – RCS Leakage 6 – RPS Failure 7 – Loss of Communications 8 – Containment Failure 9 – Hazardous Event Affecting Safety Systems	**S**ystem Malfunction ICs/EALs
F – Fission Product Barrier	None	**F**ission Product Barrier ICs/EALs
Group: Cold Conditions:		
C – Cold Shutdown/Refueling System Malfunction	1 – RCS Level 2 – Loss of Emergency AC Power 3 – RCS Temperature 4 – Loss of Vital DC Power 5 – Loss of Communications 6 - Hazardous Event Affecting Safety Systems	**C**old Shutdown./ Refueling System Malfunction ICs/EALs

Table 2 – NEI / CNP EAL Identification Cross-Reference

NEI		CNP	
IC	Example EAL	Category and Subcategory	EAL
AU1	1	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RU1.1
AU1	2	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RU1.1
AU1	3	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RU1.2
AU2	1	R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	RU2.1
AA1	1	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RA1.1
AA1	2	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RA1.2
AA1	3	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RA1.3
AA1	4	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RA1.4
AA2	1	R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	RA2.1
AA2	2	R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	RA2.2
AA2	3	R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	RA2.3
AA3	1	R – Abnormal Rad Levels / Rad Effluent, 3 – Area Radiation Levels	RA3.1
AA3	2	R – Abnormal Rad Levels / Rad Effluent, 3 – Area Radiation Levels	RA3.2
AS1	1	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RS1.1
AS1	2	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RS1.2
AS1	3	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RS1.3

NEI		CNP	
IC	Example EAL	Category and Subcategory	EAL
AS2	1	R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	RS2.1
AG1	1	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RG1.1
AG1	2	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RG1.2
AG1	3	R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	RG1.3
AG2	1	R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	RG2.1
CU1	1	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CU1.1
CU1	2	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CU1.2
CU2	1	C – Cold SD/ Refueling System Malfunction, 2 – Loss of ESF AC Power	CU2.1
CU3	1	C – Cold SD/ Refueling System Malfunction, 3 – RCS Temperature	CU3.1
CU3	2	C – Cold SD/ Refueling System Malfunction, 3 – RCS Temperature	CU3.2
CU4	1	C – Cold SD/ Refueling System Malfunction, 4 – Loss of Vital DC Power	CU4.1
CU5	1, 2, 3	C – Cold SD/ Refueling System Malfunction, 5 – Loss of Communications	CU5.1
CA1	1	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CA1.1
CA1	2	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CA1.2
CA2	1	C – Cold SD/ Refueling System Malfunction, 1 – Loss of ESF AC Power	CA2.1
CA3	1, 2	C – Cold SD/ Refueling System Malfunction, 3 – RCS Temperature	CA3.1
CA6	1	C – Cold SD/ Refueling System Malfunction, 6 – Hazardous Event Affecting Safety Systems	CA6.1
CS1	1	N/A	N/A

NEI		CNP	
IC	Example EAL	Category and Subcategory	EAL
CS1	2	N/A	N/A
CS1	3	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CS1.1
CG1	1	N/A	N/A
CG1	2	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CG1.1
E-HU1	1	E - ISFSI	EU1.1
FA1	1	F – Fission Product Barrier Degradation	FA1.1
FS1	1	F – Fission Product Barrier Degradation	FS1,1
FG1	1	F – Fission Product Barrier Degradation	FG1.1
HU1	1, 2, 3	H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security	HU1.1
HU2	1	H – Hazards and Other Conditions Affecting Plant Safety, 2 – Seismic Event	HU2.1
HU3	1	H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technological Hazard	HU3.1
HU3	2	H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technological Hazard	HU3.2
HU3	3	H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technological Hazard	HU3.3
HU3	4	H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technological Hazard	HU3.4
HU3	5	N/A	N/A
HU4	1	H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire or Explosion	HU4.1
HU4	2	H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire or Explosion	HU4.2
HU4	3	H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire or Explosion	HU4.3

NEI		CNP	
IC	Example EAL	Category and Subcategory	EAL
HU4	4	H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire or Explosion	HU4.4
HU7	1	H – Hazards and Other Conditions Affecting Plant Safety, 7 – Judgment	HU7.1
HA1	1, 2	H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security	HA1.1
HA5	1	H – Hazards and Other Conditions Affecting Plant Safety, 5 – Hazardous Gases	HA5.1
HA6	1	H – Hazards and Other Conditions Affecting Plant Safety, 6 – Control Room Evacuation	HA6.1
HA7	1	H – Hazards and Other Conditions Affecting Plant Safety, 7 – Judgment	HA7.1
HS1	1	H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security	HS1.1
HS6	1	H – Hazards and Other Conditions Affecting Plant Safety, 6 – Control Room Evacuation	HS6.1
HS7	1	H – Hazards and Other Conditions Affecting Plant Safety, 7 – Judgment	HS7.1
HG1	1	N/A	N/A
HG7	1	H – Hazards and Other Conditions Affecting Plant Safety, 7 – Judgment	HG7.1
SU1	1	S – System Malfunction, 1 – Loss of Emergency AC Power	SU1.1
SU2	1	S – System Malfunction, 3 – Loss of Control Room Indications	SU3.1
SU3	1	S – System Malfunction, 4 – RCS Activity	SU4.1
SU3	2	S – System Malfunction, 4 – RCS Activity	SU4.2
SU4	1, 2, 3	S – System Malfunction, 5 – RCS Leakage	SU5.1
SU5	1	S – System Malfunction, 6 – RPS Failure	SU6.1
SU5	2	S – System Malfunction, 6 – RPS Failure	SU6.2

NEI		CNP	
IC	Example EAL	Category and Subcategory	EAL
SU6	1, 2, 3	S – System Malfunction, 7 –Loss of Communications	SU7.1
SU7	1, 2	S – System Malfunction, 8 –Containment Failure	SU8.1
SA1	1	S – System Malfunction, 1 – Loss of Emergency AC Power	SA1.1
SA2	1	S – System Malfunction, 3 – Loss of Control Room Indications	SA3.1
SA5	1	S – System Malfunction, 6 – RPS Failure	SA6.1
SA9	1	S – Hazardous Event Affecting Safety Systems	SA9.1
SS1	1	S – System Malfunction, 1 – Loss of Emergency AC Power	SS1.1
SS5	1	S – System Malfunction, 6 – RPS Failure	SS6.1
SS8	1	S – System Malfunction, 2 – Loss of Vital DC Power	SS2.1
SG1	1	S – System Malfunction, 1 – Loss of Emergency AC Power	SG1.1
SG8	1	S – System Malfunction, 2 – Loss of Vital DC Power	SG2.1

Table 3 – Summary of Deviations

NEI IC	NEI Example EAL	CNP EAL	Description
HG1	1	N/A	Generic IC HG1 and associated example EAL are not implemented in the CNP scheme. There are several other ICs that are redundant with this IC, and are better suited to ensure timely and effective emergency declarations. In addition, the development of new spent fuel pool level EALs, as a result of NRC Order EA-12-051, clarified the intended emergency classification level for spent fuel pool level events. This deviation is justified because: 1. Hostile Action in the Protected Area is bounded by ICs HS1 and HS7. Hostile Action resulting in a loss of physical control is bound by EAL HG7, as well as any event that may lead to radiological releases to the public in excess of Environmental Protection Agency (EPA) Protective Action Guides (PAGs). a. If, for whatever reason, the Control Room must be evacuated, and control of safety functions (e.g., reactivity control, core cooling, and RCS heat removal) cannot be reestablished, then IC HS6 would apply, as well as IC HS7 if desired by the EAL decision-maker. b. Also, as stated above, any event (including Hostile Action) that could reasonably be expected to have a release exceeding EPA PAGs would be bound by IC HG7. c. From a Hostile Action perspective, ICs HS1, HS7 and HG7 are appropriate, and therefore, make this part of HG1 redundant and unnecessary. d. From a loss of physical control perspective, ICs HS6, HS7 and HG7 are appropriate, and therefore, make this part of HG1 redundant and unnecessary. 2. Any event which causes a loss of spent fuel pool level will be bounded by generic ICs AA2, AS2 and AG2, regardless of whether it was based upon a Hostile Action or not, thus making this part of HG1 redundant and unnecessary.

NEI		CNP EAL	Description
IC	Example EAL		
			a. An event that leads to a radiological release will be bounded by generic ICs AU1, AA1, AS1 and AG1. Events that lead to radiological releases in excess of EPA PAGs will be bounded by EALs AG1 and HG7, thus making this part of HG1 redundant and unnecessary. ICs AA2, AS2, AG2, AS1, AG1, HS1, HS6, HS7 and HG7 have been implemented consistent with NEI 99-01 Revision 6 and thus HG1 is adequately bounded as described above. **This is an acceptable deviation from the generic NEI 99-01 Revision 6 guidance and is consistent with NRC endorsed EP FAQ 2015-13.**
HS6	1	HS6.1	Deleted defueled mode applicability. Control of the cited safety functions are not critical for a defueled reactor as there is no energy source in the reactor vessel or RCS. The Mode applicability for the reactivity control safety function has been limited to Modes 1, 2, and 3 (hot operating conditions). In the cold operating modes adequate shutdown margin exists under all conditions. **This is an acceptable deviation from the generic NEI 99-01 Revision 6 guidance and is consistent with NRC endorsed EP FAQ 2015-14.**

Category A

Abnormal Rad Levels / Radiological Effluent

NEI IC#	NEI IC Wording and Mode Applicability	CNP IC#(s)	CNP IC Wording and Mode Applicability	Difference/Deviation Justification
AU1	Release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer. MODE: All	RU1	Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer MODE: All	The CNP ODCM is the site-specific effluent release controlling document.

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Reading on **ANY** effluent radiation monitor greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer: (site-specific monitor list and threshold values corresponding to 2 times the controlling document limits)	RU1.1	Reading on **any** Table R-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3)	Example EALs #1 and #2 have been combined into a single EAL to simplify presentation. The NEI phrase "…effluent radiation monitor greater than 2 times the (site-specific effluent release controlling document)" and "effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit " have been replaced with "…**any** Table R-1 effluent radiation monitor > column "UE". UE thresholds for all CNP continuously monitored gaseous release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL user. The values shown in Table R-1 column "UE", consistent with the NEI bases, represent two times the ODCM release limits for both liquid and gaseous release.
2	Reading on **ANY** effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.			
3	Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the (site-specific effluent release controlling document) limits for	RU1.2	Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2)	The CNP ODCM is the site-specific effluent release controlling document.

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
	60 minutes or longer.			
Notes	• The Emergency Director should declare the Unusual Event promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded. • If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 60 minutes. • If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit. Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording. The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording. None

Table R-1 Effluent Monitor Classification Thresholds						
Release Point		**Monitor**	**GE**	**SAE**	**ALERT**	**UE**
Gaseous	Unit Vent Noble Gas	VRS-1500 (2500)	3.3E+00 µCi/cc	3.3E-01 µCi/cc	3.3E-02 µCi/cc	4.2E-03 µCi/cc
	Gland Seal Leakoff	SRA-1800 (2800)	1.6E+02 µCi/cc	1.6E+01 µCi/cc	1.6E+00 µCi/cc	1.4E-01 µCi/cc
	Steam Jet Air Ejector	SRA-1900 (2900)	1.5E+04 µCi/cc	1.5E+03 µCi/cc	1.5E+02 µCi/cc	1.3E+01 µCi/cc
Liquid	Radwaste Effluent	RRS-1001	----	----	----	4.6E+04 cpm
	SG Blowdown	R-19	----	----	----	1.7E+03 cpm
		DRS-3100/4100	----	----	----	1.2E+04 cpm
	SG Blowdown Treatment	R-24	----	----	----	2.9E+04 cpm
		DRS-3200/4200	----	----	----	1.2E+05 cpm

NEI IC#	NEI IC Wording and Mode Applicability	CNP IC#(s)	CNP IC Wording and Mode Applicability	Difference/Deviation Justification
AU2	UNPLANNED loss of water level above irradiated fuel. MODE: All	RU2	Unplanned loss of water level above irradiated fuel MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following; (site-specific level indications). **AND** b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors. (site-specific list of area radiation monitors)	RU2.1	UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication **AND** UNPLANNED rise in corresponding area radiation levels as indicated by **any** of the following radiation monitors: • VRS-1101/1201, Unit 1 Upper Containment • VRS-2101/2201, Unit 2 Upper Containment • R-5 Spent Fuel Area • VRS-5006 Spent Fuel Area	Added the term "...corresponding..." to emphasize the cause and effect of the water level drop in the refueling pathway. The site-specific list of radiation monitors are listed in bullet format for ease of reading.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
AA1	Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE. MODE: All	RA1	Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	RA1.1	Reading on **any** Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4)	The CNP radiation monitors that detect radioactivity effluent release to the environment are listed in Table R-1. UE, Alert, SAE and GE thresholds for all CNP continuously monitored gaseous and liquid release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL-user.
2	Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point).	RA1.2	Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the site boundary (Note 4)	The site boundary is the site-specific receptor point.
3	Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point) for one hour of exposure.	RA1.3	Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the site boundary for 60 min. of exposure (Notes 1, 2)	The site boundary is the site-specific receptor point.

4	Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point): • Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.	RA1.4	Field survey results indicate **EITHER** of the following at or beyond the site boundary: • Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min. • Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation. (Notes 1, 2)	The site boundary is the site-specific receptor point.
Notes	• The Emergency Director should declare the Alert promptly upon determining that the applicable time has been exceeded, or will likely be exceeded. • If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes. • If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes. • The pre-calculated effluent monitor values presented in EAL #1 should be used for emergency classification	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit. Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes. Note 4 The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording. The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording. None Incorporated site-specific EAL numbers associated with generic EAL#1.

	assessments until the results from a dose assessment using actual meteorology are available.		results from a dose assessment using actual meteorology are available.	

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
AA2	Significant lowering of water level above, or damage to, irradiated fuel. MODE: All	RA2	Significant lowering of water level above, or damage to, irradiated fuel MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Uncovery of irradiated fuel in the REFUELING PATHWAY.	RA2.1	Uncovery of irradiated fuel in the REFUELING PATHWAY	None
2	Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by **ANY** of the following radiation monitors: (site-specific listing of radiation monitors, and the associated readings, setpoints and/or alarms)	RA2.2	Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by High alarm on **any** of the following radiation monitors: • VRS-1101/1201, Unit 1 Upper Containment • VRS-2101/2201, Unit 2 Upper Containment • R-5 Spent Fuel Area • VRS-5006 Spent Fuel Area	The site-specific list of radiation monitors are listed in bullet format for ease of reading. The high alarm setpoints for the radiation monitors are indicative of significant increases in area and/or airborne radiation.
3	Lowering of spent fuel pool level to (site-specific Level 2 value). [*See Developer Notes*]	RA2.3	Lowering of spent fuel pool level to 9 ft. 6 in. on 1(2)-RLI-502-CRI Spent Fuel Pit Level Indication (8 ft. 10 in. on local ruler)	Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3). For CNP SFP Level 2 is plant elevation 630 ft. 10.5 in. or 9 ft. 6 in. as indicated on 1(2)-RLI-502-CRI in the Control Room or 1(2)-RLI-502-BATT back-up indicator (ref. 2). This level corresponds to 8 ft. 10 in. on the SFP ruler.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
AA3	Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown MODE: All	RA3	Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Dose rate greater than 15 mR/hr in **ANY** of the following areas: • Control Room • Central Alarm Station • (other site-specific areas/rooms)	RA3.1	Dose rate > 15 mR/hr in **any** of the following areas: • Unit 1 Control Room (ERS-7401) • Unit 2 Control Room (ERS-8401) • Central Alarm Station (by survey) • Secondary Alarm Station (by survey)	The Secondary Alarm Station (SAS) also requires continuous occupancy at CNP. ERS-7401 and ERS-8401 monitors the Control room for area radiation. Neither CAS nor SAS have installed area radiation monitoring and thus must be determined by survey.
2	An UNPLANNED event results in radiation levels that prohibit or impede access to any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode applicability identified)	RA3.2	An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 rooms or areas (Note 5)	Table R-2 contains the site-specific list of plant rooms or areas with entry-related mode applicability identified.
Note	If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.	Note 5	If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.	None

Table R-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Auxiliary Building 573'	4, 5
Auxiliary Building 587' (including D/G room)	1, 2, 3, 4, 5
Auxiliary Building 591'	1, 2, 3, 4
Auxiliary Building 609' (including 4kV room)	1, 2, 3, 4, 5
Auxiliary Building 633'	1, 2, 3, 4
Turbine Building (**All** Levels)	1, 2, 3
Turbine Building 591'	4, 5
Screenhouse	1, 2, 3, 4, 5

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
AS1	Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE MODE: All MODE: All	RS1	Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	RS1.1	Reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4)	The CNP radiation monitors that detect radioactivity effluent release to the environment are listed in Table R-1. UE, Alert, SAE and GE thresholds for all CNP continuously monitored gaseous and liquid release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL-user.
2	Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond (site-specific dose receptor point)	RS1.2	Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the site boundary (Note 4)	The site boundary is the site-specific receptor point.
3	Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point): • Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid	RS1.3	Field survey results indicate **EITHER** of the following at or beyond the site boundary: • Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min. • Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation.	The site boundary is the site-specific receptor point.

	CDE greater than 500 mrem for one hour of inhalation.		(Notes 1, 2)	
Notes	• The Emergency Director should declare the Site Area Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.		Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.
	• If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.		Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.
	• If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.		Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.	None
	• The pre-calculated effluent monitor values presented in EAL #1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.		Note 4 The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.	Incorporated site-specific EAL numbers associated with generic EAL#1.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
AS2	Spent fuel pool level at (site-specific Level 3 description) MODE: All	RS2	Spent fuel pool level at the top of the fuel racks	Top of the fuel racks is the site-specific Level 3 description.

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Lowering of spent fuel pool level to (site-specific Level 3 value)	RS2.1	Lowering of spent fuel pool level to 0 ft. on 1(2)-RLI-502-CRI Spent Fuel Pit Indication	Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3). For CNP SFP Level 3 is plant elevation 620 ft. 10.5 in. However, the SFP level instrument lower range (0 ft.) corresponds to plant elevation 621 ft. 6 in. Therefore an indicated level of 0 ft. on 1(2)-RLI-502-CRI in the Control Room or 1(2)-RLI-502-BATT back-up indicator is used as indicated Level 3.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
AG1	Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE. MODE: All	RG1	Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	RG1.1	Reading on **any** Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4)	The CNP radiation monitors that detect radioactivity effluent release to the environment are listed in Table R-1. UE, Alert, SAE and GE thresholds for all CNP continuously monitored gaseous or liquid release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL-user.
2	Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond (site-specific dose receptor point).	RG1.2	Dose assessment using actual meteorology indicates doses > 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond the site boundary (Note 4)	The site boundary is the site-specific receptor point.
3	Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point): • Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer. • Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for	RG1.3	Field survey results indicate **EITHER** of the following at or beyond the site boundary: • Closed window dose rates > 1,000 mR/hr expected to continue for ≥ 60 min. • Analyses of field survey samples indicate thyroid CDE > 5,000 mrem for 60 min. of inhalation.	The site boundary is the site-specific receptor point.

	one hour of inhalation.		(Notes 1, 2)	
Notes	• The Emergency Director should declare the Site Area Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded. • If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes. • If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes. • The pre-calculated effluent monitor values presented in EAL #1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.		Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit. Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes. Note 4 The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording. The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording. None Incorporated site-specific EAL numbers associated with generic EAL#1.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
AG2	Spent fuel pool level cannot be restored to at least (site-specific Level 3 description) for 60 minutes or longer MODE: All	RG2	Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer	Top of the fuel racks is the site-specific Level 3 description.

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Spent fuel pool level cannot be restored to at least (site-specific Level 3 value) for 60 minutes or longer	RG2.1	Spent fuel pool level cannot be restored to at least 0 ft. on 1(2)-RLI-502-CRI Spent Fuel Pit Indication for ≥ 60 min. (Note 1)	Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3). For CNP SFP Level 3 is plant elevation 620 ft. 10.5 in. However, the SFP level instrument lower range (0 ft.) corresponds to plant elevation 621 ft. 6 in. Therefore an indicated level of 0 ft. on 1(2)-RLI-502-CRI in the Control Room or 1(2)-RLI-502-BATT back-up indicator is used as indicated Level 3.
Note	The Emergency Director should declare the General Emergency promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

Category C

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
CU1	UNPLANNED loss of (reactor vessel/RCS [*PWR*] or RCP [*BWR*]) inventory for 15 minutes or longer. MODE: Cold Shutdown, Refueling	CU1	UNPLANNED loss of RCS inventory MODE: 5 - Cold Shutdown, 6 - Refueling	Deleted "…for 15 minutes or longer" because the 15 minute timing component only applies to example EAL #1.

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	UNPLANNED loss of reactor coolant results in (reactor vessel/RCS [*PWR*] or RCP [*BWR*]) level less than a required lower limit for 15 minutes or longer.	CU1.1	UNPLANNED loss of reactor coolant results in RCS water level less than a required lower limit for ≥ 15 min. (Note 1)	None
2	a. (Reactor vessel/RCS [*PWR*] or RCP [*BWR*]) level cannot be monitored. **AND** b. UNPLANNED increase in (site-specific sump and/or tank) levels.	CU1.2	RCS water level cannot be monitored **AND EITHER** • UNPLANNED increase in any Table C-1 sump/tank level due to loss of RCS inventory • Visual observation of UNISOLABLE RCS leakage	Added the phrase "due to a loss of RCS inventory" because the NEI basis states: "Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS." Table C-2 lists the CNP sumps and tanks. Added bulleted criterion "Visual observation of UNISOLABLE RCS leakage" to include direct observation of RCS leakage.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

	likely be exceeded.		exceeded, or will likely be exceeded.	

Table C-1 Sumps / Tanks
• Containment Sumps • Auxiliary Building Sumps • RWST • RCDT

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
CU2	Loss of all but one AC power source to emergency buses for 15 minutes or longer. MODE: Cold Shutdown, Refueling, Defueled	CU2	Loss of all but one AC power source to emergency buses for 15 minutes or longer. MODE: 5 - Cold Shutdown, 6 - Refueling, D - Defueled	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. AC power capability to (site-specific emergency buses) is reduced to a single power source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS.	CU2.1	AC power capability, Table C-3, to emergency 4.16 kV buses T11A (T21A) and T11D (T21D) reduced to a single power source for ≥ 15 min. (Note 1) **AND** **Any** additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS	4.16 kV buses T11A (T21A) and T11D (T21D) are the site-specific emergency buses. Site-specific AC power sources are listed in Table C-3.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

Table C-3 AC Power Sources
Offsite: • Reserve Auxiliary Xmr TR101AB (TR201AB) • Reserve Auxiliary Xmr TR101CD (TR201CD) • 69/4.16 kV Alternate Xfr TR12EP-1 • Main Xmr TR1 (TR2) backfeed (only if already aligned) **Onsite:** • EDG 1AB (2AB) • EDG 1CD (2CD)

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
CU3	UNPLANNED increase in RCS temperature MODE: Cold Shutdown, Refueling	CU3	UNPLANNED increase in RCS temperature MODE: Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit)	CU3.1	UNPLANNED increase in RCS temperature to > 200°F due to loss of decay heat removal capability (Note 10)	200°F is the site-specific Tech. Spec. cold shutdown temperature limit. Added “due to loss of decay heat removal capability” to reinforce the generic bases that states “EAL #1 involves a loss of decay heat removal capability” Added reference to Note 10 to ensure EAL thresholds applicable to Mode 4 are observed since an RCS temperature > 200°F is a mode shift from cold conditions to Mode 4.
2	Loss of **ALL** RCS temperature and (reactor vessel/RCS [*PWR*] or RCP [*BWR*]) level indication for 15 minutes or longer.	CU3.2	Loss of **all** RCS temperature and RCS level indication for ≥ 15 min. (Note 1)	None

Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.
		N/A	Note 10: Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss of decay heat removal.	Since an RCS temperature > 200°F is a mode shift from cold conditions to Mode 4, EAL thresholds applicable to Mode 4 are candidates for emergency classification.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
CU4	Loss of Vital DC power for 15 minutes or longer. MODE: Cold Shutdown, Refueling	CU4	Loss of Vital DC power for 15 minutes or longer. MODE: 5 - Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Indicated voltage is less than (site-specific bus voltage value) on required Vital DC buses for 15 minutes or longer.	CU4.1	< 215 VDC bus voltage indications on Technical Specification **required** 250 VDC vital buses for ≥ 15 min. (Note 1)	215 VDC is the site-specific minimum vital DC bus voltage. DC operability requirements are specified in Technical Specifications.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
CU5	Loss of all onsite or offsite communications capabilities. MODE: Cold Shutdown, Refueling, Defueled	CU5	Loss of **all** onsite or offsite communications capabilities. MODE: 5 - Cold Shutdown, 6 - Refueling, D - Defueled	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Loss of **ALL** of the following onsite communication methods: (site specific list of communications methods)	CU5.1	Loss of **all** Table C-5 onsite communication methods **OR** Loss of **all** Table C-5 ORO communication methods **OR** Loss of **all** Table C-5 NRC communication methods	Example EALs #1, 2 and 3 have been combined into a single EAL for simplification. Table C-5 provides a site-specific list of onsite, ORO and NRC communications methods.
2	Loss of **ALL** of the following ORO communications methods: (site specific list of communications methods)			
3	Loss of **ALL** of the following NRC communications methods: (site specific list of communications methods)			

Table C-5 Communication Methods

System	Onsite	ORO	NRC
Plant Page	X		
Plant Radios	X	X	
Plant Telephone	X	X	X
ENS Line		X	X
Commercial Telephone		X	X
Microwave Transmission		X	X

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
CA1	Loss of (reactor vessel/RCS [*PWR*] or RCP [*BWR*]) inventory MODE: Cold Shutdown, Refueling	CA1	Significant loss of RCS inventory MODE: 5 - Cold Shutdown, 6 - Refueling	Added the word "Significant…" to differentiate IC CA1 from IC CU1.

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Loss of (reactor vessel/RCS [*PWR*] or RCP [*BWR*]) inventory as indicated by level less than (site-specific level).	CA1.1	Loss of RCS inventory as indicated by RCS level < 614.0 ft.	614.0 ft. corresponds to midloop and is the minimum allowed RCS level for operation of RHR.
2	a. (Reactor vessel/RCS [*PWR*] or RCP [*BWR*]) level cannot be monitored for 15 minutes or longer **AND** b. UNPLANNED increase in (site-specific sump and/or tank) levels due to a loss of (reactor vessel/RCS [*PWR*] or RCP [*BWR*]) inventory.	CA1.2	RCS water level cannot be monitored for ≥ 15 min. (Note 1) **AND EITHER** • UNPLANNED increase in **any** Table C-1 sump/tank level due to a loss of RCS inventory • Visual observation of UNISOLABLE RCS leakage	Added the phrase "due to a loss of RCS inventory" because the NEI basis states: "Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS." Table C-1 lists the CNP sumps and tanks. Added bulleted criterion "Visual observation of UNISOLABLE RCS leakage" to include direct observation of RCS leakage.
Note	The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
CA2	Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer MODE: Cold Shutdown, Refueling, Defueled	CA2	Loss of **all** offsite and **all** onsite AC power to emergency buses for 15 minutes or longer. MODE: 5 - Cold Shutdown, 6 - Refueling, D - Defueled	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Loss of **ALL** offsite and **ALL** onsite AC Power to (site-specific emergency buses) for 15 minutes or longer.	CA2.1	Loss of **all** offsite and **all** onsite AC power to emergency 4.16KV buses T11A (T21A) and T11D (T21D) for ≥ 15 min. (Note 1)	4.16KV buses T11A (T21A) and T11D (T21D) are the site-specific emergency buses.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
CA3	Inability to maintain the plant in cold shutdown. MODE: Cold Shutdown, Refueling	CA3	Inability to maintain the plant in cold shutdown. MODE: 5 - Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit) for greater than the duration specified in the following table.	CA3.1	UNPLANNED increase in RCS temperature to > 200°F for > Table C-4 duration (Notes 1, 10) **OR** UNPLANNED RCS pressure increase > 10 psig (This EAL does not apply during water-solid plant conditions)	Example EALs #1 and #2 have been combined into a single EAL as EAL #2 is the alternative threshold based on a loss of RCS temperature indication. 200°F is the site-specific Tech. Spec. cold shutdown temperature limit. Table C-4 is the site-specific implementation of the generic RCS Heat-up Duration Threshold table. Added reference to Note 10 to ensure EAL thresholds applicable to Mode 4 are observed since an RCS temperature > 200°F is a mode shift from cold conditions to Mode 4. 10 psig is the site-specific pressure increase readable on Control Room indications.
2	UNPLANNED RCS pressure increase greater than (site-specific pressure reading). (This EAL does not apply during water-solid plant conditions. [*PWR*])			
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. Note 10: Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording. Since an RCS temperature > 200°F is a mode shift from cold conditions to Mode 4, EAL thresholds applicable to Mode 4 are candidates for emergency classification.

			of decay heat removal.	

Table: RCS Heat-up Duration Thresholds		
RCS Status	**Containment Closure Status**	**Heat-up Duration**
Intact (but not at reduced inventory [*PWR*])	Not applicable	60 minutes*
Not intact (or at reduced inventory [*PWR*])	Established	20 minutes*
	Not Established	0 minutes
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

Table C-4: RCS Heat-up Duration Thresholds		
RCS Status	**CONTAINMENT CLOSURE Status**	**Heat-up Duration**
INTACT (but **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.		

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
CA6	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. MODE: Cold Shutdown, Refueling	CA6	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. MODE: 5 - Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. The occurrence of **ANY** of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • (site-specific hazards) • Other events with similar hazard characteristics as determined by the Shift Manager **AND** b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode. **OR** 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	CA6.1	The occurrence of **any** Table C-6 hazardous event **AND EITHER**: • Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode • The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode	The hazardous events are listed in Table C-6 for simplification and clarification.

Table C-6 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the SEC

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
CS1	Loss of (reactor vessel/RCS [*PWR*] or RCP [*BWR*]) inventory affecting core decay heat removal capability. MODE: Cold Shutdown, Refueling	CS1	Loss of RCS inventory affecting core decay heat removal capability MODE: Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. CONTAINMENT CLOSURE not established. **AND** b. (Reactor vessel/RCS [*PWR*] or RCP [*BWR*]) level less than (site-specific level).	N/A	N/A	The design and operation of water level instrumentation is such that the "site-specific level" (6" below the bottom ID of the RCS loop) cannot be determined at any time during Cold Shutdown or Refueling modes, Classification is accomplished in accordance with EAL #3.
2	a. CONTAINMENT CLOSURE established. **AND** b. (Reactor vessel/RCS [*PWR*] or RCP [*BWR*]) level less than (site-specific level).	N/A	N/A	The design and operation of water level instrumentation is such that the "site-specific level" (top of active fuel) cannot be determined at any time during Cold Shutdown or Refueling modes, Classification is accomplished in accordance with EAL #3.
3	a. (Reactor vessel/RCS [*PWR*] or RCP [*BWR*]) level cannot be monitored for 30 minutes or longer. **AND** b. Core uncovery is indicated by **ANY** of the following:	CS1.1	RCS water level cannot be monitored for ≥ 30 min. (Note 1) **AND** Core uncovery is indicated by **any** of the following: • UNPLANNED increase in **any** Table C-1 sump/tank level of sufficient	Table C-1 lists the CNP sumps and tanks. A high alarm on Containment radiation monitors VRA-1310(2310) or VRA-1410(2410) would be indicative of possible core uncovery in the Refueling mode.

	• (Site-specific radiation monitor) reading greater than (site-specific value) • Erratic source range monitor indication [*PWR*] • UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery • (Other site-specific indications)		magnitude to indicate core uncovery • High alarm on Containment radiation monitor VRA-1310 (2310) or VRA-1410(2410) • Erratic Source Range Monitor indication	
Note	The Emergency Director should declare the Site Area Emergency promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
CG1	Loss of (reactor vessel/RCS [*PWR*] or RCP [*BWR*]) inventory affecting fuel clad integrity with containment challenged MODE: Cold Shutdown, Refueling	CG1	Loss of RCS inventory affecting fuel clad integrity with containment challenged MODE: Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. (Reactor vessel/RCS [*PWR*] or RCP [*BWR*]) level less than (site-specific level) for 30 minutes or longer. **AND** b. **ANY** indication from the Containment Challenge Table (see below).	N/A	N/A	The design and operation of water level instrumentation is such that the "site-specific level" (6" below the bottom ID of the RCS loop) cannot be determined at any time during Cold Shutdown or Refueling modes, Classification is accomplished in accordance with EAL #3.
2	a. (Reactor vessel/RCS [*PWR*] or RCP [*BWR*]) level cannot be monitored for 30 minutes or longer. **AND** b. Core uncovery is indicated by **ANY** of the following: • (Site-specific radiation monitor) reading greater than (site-specific value) • Erratic source range monitor indication [*PWR*]	CG1.1	RCS water level cannot be monitored for ≥ 30 min. (Note 1) **AND** Core uncovery is indicated by **any** of the following: • UNPLANNED increase in **any** Table C-1 sump/tank level of sufficient magnitude to indicate core uncovery • High alarm on Containment radiation monitor VRA-1310 (2310) or VRA-1410(2410) • Erratic Source Range	Table C-1 lists the CNP sumps and tanks. A high alarm on Containment radiation monitors VRA-1310(2310) or VRA-1410(2410) would be indicative of possible core uncovery in the Refueling mode. Table C-2 lists containment challenge indications. The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations.

	• UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery • (Other site-specific indications) **AND** c. **ANY** indication from the Containment Challenge Table (see below).		Monitor indication **AND** **Any** Containment Challenge indication, Table C-2	
Note	The Emergency Director should declare the General Emergency promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded. N/A	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a GENERAL EMERGENCY is not required.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording. Note 6 implements the asterisked note associated with the generic Containment Challenge table.

Containment Challenge Table
■ CONTAINMENT CLOSURE not established*
■ (Explosive mixture) exists inside containment
■ UNPLANNED increase in containment pressure
■ Secondary containment radiation monitor reading above (site-specific value) [*BWR*]

* If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a GENERAL EMERGENCY is not required.

Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • Containment hydrogen concentration ≥ 4% • Unplanned rise in Containment pressure

Category D

Permanently Defueled Station Malfunction

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
PD-AU1 PD-AU2 PD-SU1 PD-HU1 PD-HU2 PD-HU3 PD-AA1 PD-AA2 PD-HA1 PD-HA3	Recognition Category D Permanently Defueled Station	N/A	N/A	NEI Recognition Category PD ICs and EALs are applicable only to permanently defueled stations. CNP is not a defueled station.

Category E

Independent Spent Fuel Storage Installation

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
E-HU1	Damage to a loaded cask CONFINEMENT BOUNDARY MODE: All	EU1	Damage to a loaded cask CONFINEMENT BOUNDARY MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than (2 times the site-specific cask specific technical specification allowable radiation level) on the surface of the spent fuel cask.	EU1.1	Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading: • > 60 mrem/hr (gamma + neutron) on the top of the overpack • > 600 mrem/hr (gamma + neutron) on the side of the overpack excluding inlet and outlet ducts	The listed radiation reading values represent 2 times the limits specified in the ISFSI Certificate of Compliance Technical Specification for radiation external to a loaded cask.

Category F

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
FA1	Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier. MODE: Power Operation, Hot Standby, Startup, Hot Shutdown	FA1	**Any** loss or **any** potential loss of **EITHER** Fuel Clad **OR** RCS MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier.	FA1.1	**Any** loss or **any** potential loss of **EITHER** Fuel Clad **OR** RCS (Table F-1)	Table F-1 provides the fission product barrier loss and potential loss thresholds.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
FS1	Loss or Potential Loss of any two barriers MODE: Power Operation, Hot Standby, Startup, Hot Shutdown	FS1	Loss or potential loss of **any** two barriers MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Loss or Potential Loss of any two barriers	FS1.1	Loss or potential loss of **any** two barriers	Table F-1 provides the fission product barrier loss and potential loss thresholds.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
FG1	Loss of any two barriers and Loss or Potential Loss of third barrier MODE: Power Operation, Hot Standby, Startup, Hot Shutdown	FG1	Loss of **any** two barriers and loss or potential loss of the third barrier MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Loss of any two barriers and Loss or Potential Loss of third barrier	FG1.1	Loss of **any** two barriers **AND** Loss or potential loss of the third barrier (Table F-1)	Table F-1 provides the fission product barrier loss and potential loss thresholds.

PWR Fuel Clad Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI Threshold Wording	CNP FPB #(s)	CNP FPB Wording	Difference/Deviation Justification
FC Loss 1	**RCS or SG Tube Leakage** Not Applicable	N/A	N/A	N/A
FC Loss 2	**Inadequate Heat Removal** A. Core exit thermocouple readings greater than (site-specific temperature value).	FC Loss B.1	CSFST Core Cooling-RED Path (F.0-2) conditions met	Consistent with the generic developers note options CSFST Core Cooling Red Path is used in lieu of CET temperatures.
FC Loss 3	**RCS Activity/CNMT Rad** A. Containment radiation monitor reading greater than (site-specific value) **OR** B. (Site-specific indications that reactor coolant activity is greater than 300 μCi/gm dose equivalent I-131)	FC Loss C.1	Containment radiation > Table F-2 column "FC Loss" on VRA-1310/1410 (2310/2410)	CHRM VRA-1310/1410 (2310/2410) are the site-specific containment high range radiation monitors. The Table F-2 values, column FC Loss represents the expected containment high range radiation monitor response based on a LOCA.
		FC Loss C.2	Dose equivalent I-131 coolant activity > 300 μCi/gm	None
FC Loss 4	**CNMT Integrity or Bypass** Not Applicable	N/A	N/A	N/A
FC Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific Fuel Clad Loss indication has been identified for CNP.
FC Loss 6	**ED Judgment** A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	FC Loss E.1	**Any** condition in the opinion of the SEC that indicates loss of the fuel clad barrier	None

NEI FPB#	NEI Threshold Wording	CNP FPB #(s)	CNP FPB Wording	Difference/Deviation Justification
FC P-Loss 1	**RCS or SG Tube Leakage** A. RCS/reactor vessel level less than (site-specific level)	N/A	N/A	See FC P-Loss B.1. The RCS level threshold is implemented as CSFST Core Cooling Orange Path conditions met.
FC P-Loss 2	**Inadequate Heat Removal** A. Core exit thermocouple readings greater than (site-specific temperature value) **OR** B. Inadequate RCS heat removal capability via steam generators as indicated by (site-specific indications).	FC P-Loss B.1	CSFST Core Cooling-ORANGE Path (F.0-2) conditions met	Consistent with the generic developers note options CSFST Core Cooling Orange Path is used in lieu of CET temperatures.
		FC P-Loss B.2	CSFST Heat Sink-RED Path (F.0-3) conditions met **AND** Heat sink is required	Consistent with the generic developers note options CSFST Heat Sink Red Path is used. The phrase "and heat sink required" was added to preclude the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP.
FC P-Loss 3	**RCS Activity/CNMT Rad** Not Applicable	N/A	N/A	N/A
FC P-Loss 4	**CNMT Integrity or Bypass** Not Applicable	N/A	N/A	N/A
FC P-Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific Fuel Clad Potential Loss indication has been identified for CNP.

NEI FPB#	NEI Threshold Wording	CNP FPB #(s)	CNP FPB Wording	Difference/Deviation Justification
FC P-Loss 6	**Emergency Director Judgment** A. Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	FC P-Loss E.1	**Any** condition in the opinion of the SEC that indicates potential loss of the fuel clad barrier	None

Table F-2 Containment Radiation – R/hr - VRA-1310 (2310) / 1410 (2410)			
Monitor	**FC Loss**	**RCS Loss**	**CNMT Potential Loss**
VRA-1310 (2310)	1,000	200	9,100
VRA-1410 (2410)	700	140	6,300

PWR RCS Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI IC Wording	CNP FPB #(s)	CNP FPB Wording	Difference/Deviation Justification
RCS Loss 1	**RCS or SG Tube Leakage** A. An automatic or manual ECCS (SI) actuation is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube RUPTURE.	RCS Loss A.1	An automatic or manual ECCS (SI) actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	None
RCS Loss 2	**Inadequate Heat Removal** Not Applicable	N/A	N/A	N/A
RCS Loss 3	**RCS Activity/CNMT Rad** A. Containment radiation monitor reading greater than (site-specific value).	RCS Loss C.1	Containment radiation > Table F-2 column "RCS Loss" on VRA-1310/1410 (2310/2410)	CHRM VRA-1310/1410 (2310/2410) are the site-specific containment high range radiation monitors. The Table F-2 values, column RCS Loss represents the expected containment high range radiation monitor response based on a LOCA.
RCS Loss 4	**CNMT Integrity or Bypass** Not Applicable	N/A	N/A	N/A
RCS Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific RCS Loss indication has been identified for CNP.
RCS Loss	**Emergency Director Judgment** A. **ANY** condition in the opinion	RCS Loss	**Any** condition in the opinion of the SEC that indicates loss	None

NEI FPB#	NEI IC Wording	CNP FPB #(s)	CNP FPB Wording	Difference/Deviation Justification
6	of the Emergency Director that indicates Loss of the RCS Barrier.	E.1	of the RCS barrier	
RCS P-Loss 1	**RCS or SG Tube Leakage** A. Operation of a standby charging (makeup) pump is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube leakage. **OR** B. RCS cooldown rate greater than (site-specific pressurized thermal shock criteria/limits defined by site-specific indications).	RCS P-Loss A.1	Operation of a standby charging pump is required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	CNP terminology is "charging pump." "(makeup)" is therefore deleted.
		RCS P-Loss A.2	CSFST Integrity-RED Path (F.0-4) conditions met	Consistent with the generic developers note options CSFST Integrity Red Path is used.
RCS P-Loss 2	**Inadequate Heat Removal** A. Inadequate RCS heat removal capability via steam generators as indicated by (site-specific indications).	RCS P-Loss B.1	CSFST Heat Sink-RED Path (F.0-3) conditions met **AND** Heat sink is required	Consistent with the generic developers note options CSFST Heat Sink Red Path is used. The phrase "and heat sink required" was added to preclude the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP.
RCS P-Loss 3	**CS Activity/CNMT Rad** Not Applicable	N/A	N/A	N/A

NEI FPB#	NEI IC Wording	CNP FPB #(s)	CNP FPB Wording	Difference/Deviation Justification
RCS P-Loss 4	**CNMT Integrity or Bypass** Not Applicable	N/A	N/A	N/A
RCS P-Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific RCS Potential Loss indication has been identified for CNP.
RCS P-Loss 6	**Emergency Director Judgment** A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	RCS P-Loss E.1	**Any** condition in the opinion of the SEC that indicates potential loss of the RCS barrier	None

PWR Containment Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI IC Wording	CNP FPB #(s)	CNP FPB Wording	Difference/Deviation Justification
CNMT Loss 1	**RCS or SG Tube Leakage** A. A leaking or RUPTURED SG is FAULTED outside of containment.	CNMT Loss A.1	A leaking or RUPTURED SG is FAULTED outside of containment	None
CNMT Loss 2	**Inadequate Heat Removal** Not Applicable	N/A	N/A	N/A
CNMT Loss 3	**RCS Activity/CNMT Rad** Not applicable	N/A	N/A	N/A
CNMT Loss 4	**CNMT Integrity or Bypass** A. Containment isolation is required **AND** **EITHER** of the following: 1. Containment integrity has been lost based on Emergency Director judgment. **OR** 2. UNISOLABLE pathway from the containment to the environment exists. **OR** B. Indications of RCS leakage outside of containment.	CNMT Loss D.1	Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on SEC judgment • UNISOLABLE pathway from containment to the environment exists	None
		CNMT Loss D.2	Indications of RCS leakage outside of containment	None

NEI FPB#	NEI IC Wording	CNP FPB #(s)	CNP FPB Wording	Difference/Deviation Justification
CNMT Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific Containment Loss indication has been identified for CNP.
CNMT Loss 6	**Emergency Director Judgment** **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	CNMT Loss E.1	**Any** condition in the opinion of the SEC that indicates loss of the containment barrier	None
CNMT P-Loss 1	**RCS or SG Tube Leakage** Not Applicable	N/A	N/A	N/A
CNMT P-Loss 2	**Inadequate Heat Removal** A. 1. (Site-specific criteria for entry into core cooling restoration procedure) **AND** 2. Restoration procedure not effective within 15 minutes.	CNMT P-Loss B.1	CSFST Core Cooling-RED Path (F.0-5) conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)	Consistent with the generic developers note options CSFST Core Cooling Red Path is used in lieu of CET temperatures and RCS levels. Added Note 1 consistent with other thresholds with a timing component.
CNMT P-Loss 3	**RCS Activity/CNMT Rad** A. Containment radiation monitor reading greater than (site-specific value).	CNMT P-Loss C.1	Containment radiation > Table F-2 column "CNMT Potential Loss" on VRA-1310/1410 (2310/2410)	CHRM VRA-1310/1410 (2310/2410) are the site-specific containment high range radiation monitors. The Table F-2 values, column CNMT Potential Loss represents the expected containment high range radiation monitor response based on a LOCA with ~20% fuel failure.
CNMT P-Loss 4	**CNMT Integrity or Bypass** A. Containment pressure greater than (site-specific value)	CNMT P-Loss D.1	CSFST Containment-RED Path conditions me	Consistent with the generic developers note options CSFST Containment Red Path is used in lieu of containment pressure.

NEI FPB#	NEI IC Wording	CNP FPB #(s)	CNP FPB Wording	Difference/Deviation Justification
	OR B. Explosive mixture exists inside containment **OR** C. 1. Containment pressure greater than (site-specific pressure setpoint) **AND** 2. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer.	CNMT P-Loss D.2	Containment hydrogen concentration ≥ 4%	The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations.
		CNMT P-Loss D.3	Containment pressure > 2.8 psig with < one full train of depressurization equipment operating per design for ≥ 15 min. (Note 1, 9)	The containment pressure setpoint (2.8 psig) is the pressure at which the containment depressurization equipment should receive an actuation signal and begin performing its function per system design. Added Note 1 consistent with other thresholds with a timing component. Added Note 9 to clarify the composition of one full train of depressurization equipment.
CNMT P-Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific Containment Potential Loss indication has been identified for CNP.
CNMT P-Loss 6	**Emergency Director Judgment** A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.	CNMT P-Loss E.1	**Any** condition in the opinion of the SEC that indicates potential loss of the containment barrier	None

Category H

Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
HU1	Confirmed SECURITY CONDITION or threat MODE: All	HU1	Confirmed SECURITY CONDITION or threat. MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the (site-specific security shift supervision).	HU1.1	A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervisor **OR** Notification of a credible security threat directed at the site **OR** A validated notification from the NRC providing information of an aircraft threat	Example EALs #1, 2 and 3 have been combined into a single EAL. The Security Shift Supervisor is the site-specific Security Shift Supervision.
2	Notification of a credible security threat directed at the site.			
3	A validated notification from the NRC providing information of an aircraft threat.			

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
HU2	Seismic event greater than OBE level MODE: All	HU2	Seismic event greater than OBE level MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Seismic event greater than Operating Basis Earthquake (OBE) as indicated by: (site-specific indication that a seismic event met or exceeded OBE limits)	HU2.1	Control Room personnel feel an actual or potential seismic event **AND** The occurrence of a seismic event is confirmed in manner deemed appropriate by the Shift Manager	1(2)-OHP-4022-001-007, Earthquake, provides the guidance for determining if the OBE earthquake threshold is exceeded and any response actions are required. Because CNP seismic instrumentation does not provide direct and timely indications of having exceeded the OBE ground acceleration, the alternative EAL wording specified in the NEI HU2 developers note is implemented.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
HU3	Hazardous event. MODE: All	HU3	Hazardous event MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	A tornado strike within the PROTECTED AREA.	HU3.1	A tornado strike within the PROTECTED AREA	None
2	Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.	HU3.2	Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode	None
3	Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).	HU3.3	Movement of personnel within the plant PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)	Added the word "plant" to differentiate from the ISFSI Protected Area.
4	A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.	HU3.4	A hazardous event that results in onsite conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)	Added reference to Note 7.
5	(Site-specific list of natural or technological hazard events)	N/A	N/A	No other site-specific hazard has been identified for CNP.
Note	EAL #3 does not apply to routine traffic impediments such as fog,	N/A	Note 7: This EAL does not apply to routine traffic	This note, designated Note #7, is intended to apply to generic example EAL #4, not #3 as specified in the generic guidance.

	snow, ice, or vehicle breakdowns or accidents.		impediments such as fog, snow, ice, or vehicle breakdowns or accidents.	

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
HU4	FIRE potentially degrading the level of safety of the plant. MODE: All	HU4	FIRE potentially degrading the level of safety of the plant MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. A FIRE is NOT extinguished within 15-minutes of **ANY** of the following FIRE detection indications: • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms or indications • Field verification of a single fire alarm **AND** b. The FIRE is located within **ANY** of the following plant rooms or areas: (site-specific list of plant rooms or areas)	HU4.1	A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1): • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms or indications • Field verification of a single fire alarm **AND** The FIRE is located within **any** Table H-1 area	Table H-1 lists the site-specific fire areas.
2	a. Receipt of a single fire alarm (i.e., no other indications of a FIRE). **AND** b. The FIRE is located within	HU4.2	Receipt of a single fire alarm (i.e., **no** other indications of a FIRE) **AND** The fire alarm is indicating a	Table H-1 provides a list of site-specific fire areas.

	ANY of the following plant rooms or areas: (site-specific list of plant rooms or areas) **AND** c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.		FIRE within **any** Table H-1 area **AND** The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1)	
3	A FIRE within the plant *or ISFSI [for plants with an ISFSI outside the plant Protected Area]* PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.	HU4.3	A FIRE within the PROTECTED AREA (plant or ISFSI) **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)	CNP has an ISFSI located outside the plant Protected Area.
4	A FIRE within the plant *or ISFSI [for plants with an ISFSI outside the plant Protected Area]* PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.	HU4.4	A FIRE within the PROTECTED AREA (plant or ISFSI) that requires firefighting support by an offsite fire response agency to extinguish	CNP has an ISFSI located outside the plant Protected Area.
Note	**Note:** The Emergency Director should declare the Unusual Event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

Table H-1 Fire Areas
• Control Room • Containment • Auxiliary Building • Switchgear Areas • Diesel Generator Rooms • ESW System Enclosures • AFW Pump Rooms • Refueling Water Storage Tank • Condensate Storage Tank

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
HU7	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO)UE MODE: All	HU7	Other conditions existing that in the judgment of the SEC warrant declaration of a UE MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.	HU7.1	Other conditions exist which in the judgment of the SEC indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.	None

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
HA1	HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes. MODE: All	HA1	HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the (site-specific security shift supervision).	HA1.1	A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor **OR** A validated notification from NRC of an aircraft attack threat within 30 min. of the site	Example EALs #1 and #2 have been combined into a single EAL. The Security Shift Supervisor is the site-specific Security Shift Supervision.
2	A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.			

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
HA5	Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown. MODE: All	N/A	Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown. MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. Release of a toxic, corrosive, asphyxiant or flammable gas into any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode applicability identified) **AND** b. Entry into the room or area is prohibited or impeded.	HA5.1	Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 rooms or areas **AND** Entry into the room or area is prohibited or IMPEDED (Note 5)	Table H-2 provides a list of safe shutdown rooms/areas and applicable operating modes.
Note	**Note:** If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.	Note 5	If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.	None

Table H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Auxiliary Building 573'	4, 5
Auxiliary Building 587' (including D/G room)	1, 2, 3, 4, 5
Auxiliary Building 591'	1, 2, 3, 4
Auxiliary Building 609' (including 4kV room)	1, 2, 3, 4, 5
Auxiliary Building 633'	1, 2, 3, 4
Turbine Building (**All** Levels)	1, 2, 3
Turbine Building 591'	4, 5
Screenhouse	1, 2, 3, 4, 5

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
HA6	Control Room evacuation resulting in transfer of plant control to alternate locations. MODE: All	HA6	Control Room evacuation resulting in transfer of plant control to alternate locations MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations).	HA6.1	An event has resulted in plant control being transferred from the Control Room to the Local Shutdown Instrumentation	The Local Shutdown Instrumentation is the site-specific remote shutdown panels/local control stations.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
HA7	Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert. MODE: All	HA7	Other conditions exist that in the judgment of the SEC warrant declaration of an ALERT MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	HA7.1	Other conditions exist which, in the judgment of the SEC, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	None

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
HS1	HOSTILE ACTION within the PROTECTED AREA MODE: All	HS1	HOSTILE ACTION within the plant PROTECTED AREA MODE: All	Added the word "plant" to "Protected Area" to reinforce the generic bases that states "This IC does not apply to a HOSTILE ACTION directed at an ISFSI PROTECTED AREA located outside the plant PROTECTED AREA; such an attack should be assessed using IC HA1."

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision).	HS1.1	A HOSTILE ACTION is occurring or has occurred within the plant PROTECTED AREA as reported by the Security Shift Supervisor	The Security Shift Supervisor is the site-specific security shift supervision. Added the word "plant" to "Protected Area" to reinforce the generic bases that states "This IC does not apply to a HOSTILE ACTION directed at an ISFSI PROTECTED AREA located outside the plant PROTECTED AREA; such an attack should be assessed using IC HA1."

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
HS6	Inability to control a key safety function from outside the Control Room. MODE: All	HS6	Inability to control a key safety function from outside the Control Room MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown, 5 – Cold Shutdown, 6 - Refueling	Deleted defueled mode applicability. Control of the cited safety functions are not critical for a defueled reactor as there is no energy source in the reactor vessel or RCS. **This is an acceptable deviation from the generic NEI 99-01 Revision 6 guidance and is consistent with NRC endorsed EP FAQ 2015-14.**

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations). **AND** b. Control of **ANY** of the following key safety functions is not reestablished within (site-specific number of minutes). • Reactivity control • Core cooling [*PWR*] / RCP water level [*BWR*] • RCS heat removal	HS6.1	An event has resulted in plant control being transferred from the Control Room to the Local Shutdown Instrumentation **AND** Control of **any** of the following key safety functions is **not** reestablished within 15 min. (Note 1): • Reactivity control (modes 1. 2 and 3 **only**) • Core cooling • RCS heat removal	The Local Shutdown Instrumentation is the site-specific remote shutdown panels/local control stations. The Mode applicability for the reactivity control safety function has been limited to Modes 1, 2, and 3 (hot operating conditions). In the cold operating modes adequate shutdown margin exists under all conditions. **This is an acceptable deviation from the generic NEI 99-01 Revision 6 guidance and is consistent with NRC endorsed EP FAQ 2015-14.**

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
HS7	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency. MODE: All	HS7	Other conditions existing that in the judgment of the SEC warrant declaration of a SITE AREA EMERGENCY MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	HS7.1	Other conditions exist which in the judgment of the SEC indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	None

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
HG1	HOSTILE ACTION resulting in loss of physical control of the facility. MODE: All	N/A	N/A	IC HG1 and associated example EAL are not implemented in the CNP scheme. There are several other ICs that are redundant with this IC, and are better suited to ensure timely and effective emergency declarations. In addition, the development of new spent fuel pool level EALs, as a result of NRC Order EA-12-051, clarified the intended emergency classification level for spent fuel pool level events. **This is an acceptable deviation from the generic NEI 99-01 Revision 6 guidance and is consistent with NRC endorsed EP FAQ 2015-13.**

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision). **AND** b. **EITHER** of the following has occurred: 1. **ANY** of the following safety functions cannot be controlled or maintained. • Reactivity control • Core cooling [*PWR*]/RCP water level [*BWR*]	N/A	N/A	IC HG1 and associated example EAL are not implemented in the CNP scheme. There are several other ICs that are redundant with this IC, and are better suited to ensure timely and effective emergency declarations. In addition, the development of new spent fuel pool level EALs, as a result of NRC Order EA-12-051, clarified the intended emergency classification level for spent fuel pool level events. This deviation is justified because: 1. Hostile Action in the Protected Area is bounded by ICs HS1 and HS7. Hostile Action resulting in a loss of physical control is bound by EAL HG7, as well as any event that may lead to radiological releases to the public in excess of Environmental Protection Agency (EPA) Protective Action Guides (PAGs). a. If, for whatever reason, the Control Room must be evacuated, and control of safety functions (e.g.,

	• RCS heat removal **OR** 2. Damage to spent fuel has occurred or is IMMINENT.			reactivity control, core cooling, and RCS heat removal) cannot be reestablished, then IC HS6 would apply, as well as IC HS7 if desired by the EAL decision-maker. b. Also, as stated above, any event (including Hostile Action) that could reasonably be expected to have a release exceeding EPA PAGs would be bound by IC HG7. c. From a Hostile Action perspective, ICs HS1, HS7 and HG7 are appropriate, and therefore, make this part of HG1 redundant and unnecessary. d. From a loss of physical control perspective, ICs HS6, HS7 and HG7 are appropriate, and therefore, make this part of HG1 redundant and unnecessary. 2. Any event which causes a loss of spent fuel pool level will be bounded by ICs AA2, AS2 and AG2, regardless of whether it was based upon a Hostile Action or not, thus making this part of HG1 redundant and unnecessary. a. An event that leads to a radiological release will be bounded by ICs AU1, AA1, AS1 and AG1. Events that lead to radiological releases in excess of EPA PAGs will be bounded by EALs AG1 and HG7, thus making this part of HG1 redundant and unnecessary. ICs AA2, AS2, AG2, AS1, AG1, HS1, HS6, HS7 and HG7 have been implemented consistent with NEI 99-01 Revision 6 and thus HG1 is adequately bounded as described above. **This is an acceptable deviation from the generic NEI 99-01 Revision 6 guidance and is consistent with NRC endorsed EP FAQ 2015-13.**

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
HG7	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency MODE: All	HG7	Other conditions exist which in the judgment of the SEC warrant declaration of a GENERAL EMERGENCY MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	HG7.1	Other conditions exist which in the judgment of the SEC indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	None

Category S

System Malfunction

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SU1	Loss of all offsite AC power capability to emergency buses for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU1	Loss of **all** offsite AC power capability to emergency buses for 15 minutes or longer MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Loss of **ALL** offsite AC power capability to (site-specific emergency buses) for 15 minutes or longer.	SU1.1	Loss of **all** offsite AC power capability, Table S-1, to emergency 4.16KV buses T11A (T21A) and T11D (T21D) for ≥ 15 min. (Note 1)	4.16KV buses T11A (T21A) and T11D (T21D) are the site-specific emergency buses. Site-specific AC power sources are listed in Table S-1.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

Table S-1 AC Power Sources
Offsite: • Unit Auxiliary Xmr TR1AB (TR2AB) • Unit Auxiliary Xmr TR1CD (TR2CD) • Reserve Auxiliary Xmr TR101AB (TR201AB) • Reserve Auxiliary Xmr TR101CD (TR201CD) • 69/4.16 kV Alternate Xmr TR12EP-1 **Onsite:** • EDG 1AB (2AB) • EDG 1CD (2CD)

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SU2	UNPLANNED loss of Control Room indications for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU3	UNPLANNED loss of Control Room indications for 15 minutes or longer. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.	SU3.1	An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)	The site-specific Safety System Parameters are listed in Table S-2. CNP uses the term "Auxiliary feed flow"
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

[BWR parameter list]	*[PWR parameter list]*
Reactor Power	Reactor Power
RCP Water Level	RCS Level
RCP Pressure	RCS Pressure
Primary Containment Pressure	In-Core/Core Exit Temperature
Suppression Pool Level	Levels in at least (site-specific number) steam generators
Suppression Pool Temperature	Steam Generator Auxiliary or Emergency Feed Water Flow

Table S-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core Exit TC temperature • Level in at least one SG • Auxiliary feed flow in at least one SG

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SU3	Reactor coolant activity greater than Technical Specification allowable limits. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU4	RCS activity greater than Technical Specification allowable limits MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	Changed 'reactor coolant activity" to "RCS activity" to conform to site-specific terminology.

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	(Site-specific radiation monitor) reading greater than (site-specific value).	N/A	N/A	CNP does not have any site-specific radiation monitor that can provide readings that correspond to TS coolant activity limits.
2	Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications.	SU4.1	Sample analysis indicates RCS activity > Technical Specification Section 3.4.16 limits	Changed 'reactor coolant activity" to "RCS activity" to conform to site-specific terminology. CNP T.S. Section 3.4.16 provides the TS allowable coolant activity limits.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SU4	RCS leakage for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU5	RCS leakage for 15 minutes or longer MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	RCS unidentified or pressure boundary leakage greater than (site-specific value) for 15 minutes or longer.	SU5.1	RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. **OR** RCS identified leakage > 25 gpm for ≥ 15 min. **OR** Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min. (Note 1)	Example EALs #1, 2 and 3 have been combined into a single EAL.
2	RCS identified leakage greater than (site-specific value) for 15 minutes or longer.			
3	Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer.			
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SU5	Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor. MODE: Power Operation	SU6	Automatic or manual trip fails to shut down the reactor MODE: 1 - Power Operation	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. An automatic (trip [PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor.	SU6.1	An automatic trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** RPS setpoint is exceeded **AND** A subsequent automatic trip or manual trip action taken at the reactor control console (reactor trip switches) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)	As specified in the generic developers guidance "Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level)." Reactor power < 5% is the site-specific indication of a successful reactor trip. Added the words "... as indicated by reactor power ≥ 5% after **any** RPS setpoint is exceeded" to clarify that it is a failure of the automatic trip when a valid trip signal setpoint has been exceed. The words "subsequent automatic trip" were added to address the condition where an automatic trip signal other than the initial automatic trip failure successfully shuts down the reactor prior to any manual trip action being initiated. For example, if the reactor receives a valid reactor trip signal on high pressurizer pressure but fails to trip, but AMSAC automatically initiates and successfully trips the reactor before the manual trip signal was inserted, the EAL will still have been exceeded and an Unusual Event declared. Manual reactor trip switches are the site-specific reactor control console trip switches credited for a successful manual trip. Reactor control console is singular for CNP.
2	a. A manual trip ([PWR] / scram [BWR]) did not shutdown the reactor. **AND**	SU6.2	A manual trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** manual trip action was initiated	As specified in the generic developers guidance "Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level)." Reactor power < 5% is the site-specific indication of a

	b. **EITHER** of the following: 1. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor. **OR** 2 A subsequent automatic (trip [PWR] / scram [BWR]) is successful in shutting down the reactor.		**AND** A subsequent automatic trip or manual trip action taken at the reactor control console (reactor trip switches) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)	successful reactor trip. Added the words "... as indicated by reactor power ≥ 5% after **any** manual trip action was initiated" to clarify that it is a failure of any manual trip when an actual manual trip signal has been inserted. Combined conditions b.1 and b.2 into a single statement to simplify the presentation. Manual reactor trip switches are the site-specific reactor control console trip switches credited for a successful manual trip. Reactor control console is singular for CNP.
Notes	**Note:** A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.	N/A	Note 8: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.	None

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SU6	Loss of all onsite or offsite communications capabilities. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU7	Loss of **all** onsite or offsite communications capabilities. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Loss of **ALL** of the following onsite communication methods: (site-specific list of communications methods)	SU7.1	Loss of **all** Table S-4 onsite communication methods **OR** Loss of **all** Table S-4 ORO communication methods **OR** Loss of **all** Table S-4 NRC communication methods	Example EALs #1, 2 and 3 have been combined into a single EAL. Table S-4 provides a site-specific list of onsite, ORO and NRC communications methods.
2	Loss of **ALL** of the following ORO communications methods: (site-specific list of communications methods)			
3	Loss of **ALL** of the following NRC communications methods: (site-specific list of communications methods)			

Table S-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Plant Page	X		
Plant Radios	X	X	
Plant Telephone	X	X	X
ENS Line		X	X
Commercial Telephone		X	X
Microwave Transmission		X	X

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SU7	Failure to isolate containment or loss of containment pressure control. [*PWR*] MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU8	Failure to isolate containment or loss of containment pressure control MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. Failure of containment to isolate when required by an actuation signal. **AND** b. **ALL** required penetrations are not closed within 15 minutes of the actuation signal.	SU8.1	**Any** penetration is **not** isolated within 15 min. of a VALID containment isolation signal **OR** Containment pressure > 2.8 psig with < one full train of containment depressurization equipment operating per design for ≥ 15 min. (Note 9) (Note 1)	Reworded EAL to better describe the intent. Penetrations cannot close, but they can be isolated by closure of one or more isolation valves associated with that penetration. The revised wording maintains the generic example EAL intent while more clearly describing failure to isolate threshold. The containment pressure setpoint (2.8 psig) is the pressure at which the containment depressurization equipment should receive an actuation signal and begin performing its function per system design.
2	a. Containment pressure greater than (site-specific pressure). **AND** b. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer.			
N/A	N/A	N/A	Note 1: The SEC should declare the event promptly upon determining that time	Added Note 1 to be consistent in its use for EAL thresholds with a timing component.

			limit has been exceeded, or will likely be exceeded.	

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SA1	Loss of all but one AC power source to emergency buses for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SA1	Loss of **all but one** AC power source to emergency buses for 15 minutes or longer. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. AC power capability to (site-specific emergency buses) is reduced to a single power source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in a loss of all AC power to SAFETY SYSTEMS.	SA1.1	AC power capability, Table S-1, to emergency 4.16KV buses T11A (T21A) and T11D (T21D) reduced to a single power source for ≥ 15 min. (Note 1) **AND** **Any** additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS	4.16KV buses T11A (T21A) and T11D (T21D) are the site-specific emergency buses. Site-specific AC power sources are listed in Table S-1.
Note	The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

Table S-1 AC Power Sources
Offsite: • Unit Auxiliary Xmr TR1AB (TR2AB) • Unit Auxiliary Xmr TR1CD (TR2CD) • Reserve Auxiliary Xmr TR101AB (TR201AB) • Reserve Auxiliary Xmr TR101CD (TR201CD) • 69/4.16 kV Alternate Xmr TR12EP-1 **Onsite:** • EDG 1AB (2AB) • EDG 1CD (2CD)

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SA2	UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SA3	UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer. **AND** **ANY** of the following transient events in progress. • Automatic or manual runback greater than 25% thermal reactor power • Electrical load rejection greater than 25% full electrical load • Reactor scram [*BWR*] / trip [*PWR*] • ECCS (SI) actuation • Thermal power oscillations greater than 10% [*BWR*]	SA3.1	An UNPLANNED event results in the inability to monitor **one or more** Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1) **AND** **Any** significant transient is in progress, Table S-3	The site-specific Safety System Parameters are listed in Table S-2. CNP uses the term "Auxiliary feed flow" The site-specific significant transients are listed in Table S-3. CNP is a PWR and thus does not include thermal power oscillations > 10%.

Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

[BWR parameter list]	*[PWR parameter list]*
Reactor Power	Reactor Power
RCP Water Level	RCS Level
RCP Pressure	RCS Pressure
Primary Containment Pressure	In-Core/Core Exit Temperature
Suppression Pool Level	Levels in at least (site-specific number) steam generators
Suppression Pool Temperature	Steam Generator Auxiliary or Emergency Feed Water Flow

Table S-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core Exit TC temperature • Level in at least one SG • Auxiliary feed flow in at least one SG

Table S-3 Significant Transients
• Reactor trip • Runback ≥ 25% thermal power • Electrical load rejection > 25% of full electrical load • ECCS actuation

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SA5	Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor. MODE: Power Operation	SA6	Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control console are not successful in shutting down the reactor MODE: 1 - Power Operation	Reactor control console is singular for CNP.

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. Manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	SA6.1	An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5% **AND** Manual trip actions taken at the reactor control console (reactor trip switches) are not successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8)	As specified in the generic developers guidance "Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level)." Reactor power < 5% is the site-specific indication of a successful reactor trip. The manual reactor trip switches are the site-specific reactor control console trip switches credited for a successful manual trip.
Notes	**Note:** A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.	N/A	Note 8: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron	None

			injection strategies.	

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SA9	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SA9	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. The occurrence of **ANY** of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • (site-specific hazards) • Other events with similar hazard characteristics as determined by the Shift Manager **AND** b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode. **OR** 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	SA9.1	The occurrence of **any** Table S-5 hazardous event **AND EITHER:** • Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode • The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode	The hazardous events have been listed in Table S-5.

Table S-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the SEC

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SS1	Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SS1	Loss of **all** offsite and **all** onsite AC power to emergency buses for 15 minutes or longer MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Loss of **ALL** offsite and **ALL** onsite AC power to (site-specific emergency buses) for 15 minutes or longer.	SS1.1	Loss of **all** offsite and **all** onsite AC power to emergency 4.16KV buses T11A (T21A) and T11D (T21D) for ≥ 15 min. (Note 1)	4.16KV buses T11A (T21A) and T11D (T21D) are the site-specific emergency buses.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SS5	Inability to shutdown the reactor causing a challenge to (core cooling [*PWR*] / RCP water level [*BWR*]) or RCS heat removal. MODE: Power Operation	SS6	Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal MODE: 1 - Power Operation	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. All manual actions to shutdown the reactor have been unsuccessful. **AND** c. **EITHER** of the following conditions exist: • (Site-specific indication of an inability to adequately remove heat from the core) • (Site-specific indication of an inability to adequately remove heat from the RCS)	SS6.1	An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5% **AND** **All** actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5% **AND EITHER:** • CSFST Core Cooling RED Path (F.0-2) conditions met • CSFST Heat Sink RED Path (F.0-3) conditions met	As specified in the generic developers guidance "Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level)." Reactor power < 5% is the site-specific indication of a successful reactor trip. Indication that core cooling is extremely challenged is manifested by CSFST Core Cooling RED Path conditions met. Indication that heat removal is extremely challenged is manifested by CSFST Heat Sink RED Path conditions met.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SS8	Loss of all Vital DC power for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SS2	Loss of **all** vital DC power for 15 minutes or longer. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	Indicated voltage is less than (site-specific bus voltage value) on **ALL** (site-specific Vital DC busses) for 15 minutes or longer.	SS2.1	Loss of **all** 250 VDC power based on bus voltage indications < 215 VDC on **all** vital DC buses 1CD (2CD) (Train A) and 1AB (2AB) (Train B) for ≥ 15 min. (Note 1)	215 VDC is the site-specific minimum vital DC bus voltage. DC buses 1CD (2CD) (Train A) and 1AB (2AB) (Train B) are the site-specific vital DC buses.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SG1	Prolonged loss of all offsite and all onsite AC power to emergency buses. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SG1	Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	NEI ICs SG1 and SG8 are grouped under the same CNP IC for simplification. The CNP emergency buses are the site-specific emergency buses.

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. Loss of **ALL** offsite and **ALL** onsite AC power to (site-specific emergency buses). **AND** b. **EITHER** of the following: • Restoration of at least one AC emergency bus in less than (site-specific hours) is not likely. • (Site-specific indication of an inability to adequately remove heat from the core)	SG1.1	Loss of **all** offsite and **all** onsite AC power to emergency 4160V buses 4.16KV buses T11A (T21A) and T11D (T21D) **AND EITHER:** • Restoration of at least one emergency bus in < 4 hours is **not** likely (Note 1) • CSFST Core Cooling RED Path (F.0-2) conditions met	4.16KV buses T11A (T21A) and T11D (T21D) are the site-specific emergency buses. 4 hours is the site-specific SBO coping analysis time. CSFST Core Cooling RED Path conditions met indicates significant core exit superheating and core uncovery.
Note	The Emergency Director should declare the General Emergency promptly upon determining that (site-specific hours) has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	CNP IC#(s)	CNP IC Wording	Difference/Deviation Justification
SG8	Loss of all AC and Vital DC power sources for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SG2	Loss of **all** AC and vital DC power sources for 15 minutes or longer MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	NEI ICs SS8 and SG8 are grouped under the same CNP category.

NEI Ex. EAL #	NEI Example EAL Wording	CNP EAL #	CNP EAL Wording	Difference/Deviation Justification
1	a. Loss of **ALL** offsite and **ALL** onsite AC power to (site-specific emergency buses) for 15 minutes or longer. **AND** b. Indicated voltage is less than (site-specific bus voltage value) on ALL (site-specific Vital DC busses) for 15 minutes or longer.	SG2.1	Loss of **all** offsite and **all** onsite AC power to emergency 4160V buses T11A (T21A) and T11D (T21D) for ≥ 15 min. **AND** Loss of **all** 250 VDC power based on bus voltage indications < 215 VDC on **all** vital DC buses 1CD (2CD) (Train A) and 1AB (2AB) (Train B) for ≥ 15 min. (Note 1)	T11A (T21A) and T11D (T21D) are the site-specific emergency buses. 215 VDC is the site-specific minimum vital DC bus voltage. 1CD (2CD) (Train A) and 1AB (2AB) (Train B) are the site-specific vital DC buses.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness note has been standardized across the CNP EAL scheme by referencing the "time limit" specified within the EAL wording.

Enclosure 6 to AEP-NRC-2017-02

DONALD C. COOK NUCLEAR PLANT EMERGENCY PLAN WALL BOARDS

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

R — Abnorm. Rad Levels / Rad Effluent

1 Rad Effluent

General Emergency — RG1 Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

1 2 3 4 5 6 D

RG1.1
Reading on **any** Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4)

RG1.2
Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the site boundary (Note 4)

RG1.3
Field survey results indicate **EITHER** of the following at or beyond the site boundary:
- Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation.

(Notes 1, 2)

Site Area Emergency — RS1 Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE

1 2 3 4 5 6 D

RS1.1
Reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4)

RS1.2
Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the site boundary (Note 4)

RS1.3
Field survey results indicate **EITHER** of the following at or beyond the site boundary:
- Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation.

(Notes 1, 2)

Alert — RA1 Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

1 2 3 4 5 6 D

RA1.1
Reading on **any** Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4)

RA1.2
Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the site boundary (Note 4)

RA1.3
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the site boundary for 60 min. of exposure (Notes 1, 2)

RA1.4
Field survey results indicate **EITHER** of the following at or beyond the site boundary:
- Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation.

(Notes 1, 2)

Unusual Event — RU1 Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer

1 2 3 4 5 6 D

RU1.1
Reading on **any** Table R-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3)

RU1.2
Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2)

2 Irradiated Fuel Event

General Emergency — RG2 Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer

1 2 3 4 5 6 D

RG2.1
Spent fuel pool level **cannot** be restored to at least 0 ft. on 1(2)-RLI-502-CRI Spent Fuel Pit Level Indication for ≥ 60 min. (Note 1)

Site Area Emergency — RS2 Spent fuel pool level at the top of the fuel racks

1 2 3 4 5 6 D

RS2.1
Lowering of spent fuel pool level to 0 ft. on 1(2)-RLI-502-CRI Spent Fuel Pit Level Indication

Alert — RA2 Significant lowering of water level above, or damage to, irradiated fuel

1 2 3 4 5 6 D

RA2.1
Uncovery of irradiated fuel in the REFUELING PATHWAY

RA2.2
Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by High alarm on any of the following radiation monitors:
- VRS-1101/1201, Unit 1 Upper Containment
- VRS-2101/2201, Unit 2 Upper Containment
- R-5 Spent Fuel Area
- VRS-5006 Spent Fuel Area

RA2.3
Lowering of spent fuel pool level to 9 ft. 6 in. on 1(2)-RLI-502-CRI Spent Fuel Pit Level Indication (8 ft. 10 in. on local ruler)

Unusual Event — RU2 UNPLANNED loss of water level above irradiated fuel

1 2 3 4 5 6 D

RU2.1
UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication
AND
UNPLANNED rise in corresponding area radiation levels as indicated by **any** of the following radiation monitors:
- VRS-1101/1201, Unit 1 Upper Containment
- VRS-2101/2201, Unit 2 Upper Containment
- R-5 Spent Fuel Area
- VRS-5006 Spent Fuel Area

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Unit Vent Noble Gas	VRS-1500 (2500)	3.3E+00 μCi/cc	3.3E-01 μCi/cc	3.3E-02 μCi/cc	4.2E-03 μCi/cc
Gaseous	Gland Seal Leakoff	SRA-1800 (2800)	1.6E+02 μCi/cc	1.6E+01 μCi/cc	1.6E+00 μCi/cc	1.4E-01 μCi/cc
Gaseous	Steam Jet Air Ejector	SRA-1900 (2900)	1.5E+04 μCi/cc	1.5E+03 μCi/cc	1.5E+02 μCi/cc	1.3E+01 μCi/cc
Liquid	Radwaste Effluent	RRS-1001	—	—	—	4.6E+04 cpm
Liquid	SG Blowdown	R-19	—	—	—	1.7E+03 cpm
Liquid	SG Blowdown	DRS-3100/4100	—	—	—	1.2E+04 cpm
Liquid	SG Blowdown Treatment	R-24	—	—	—	2.9E+04 cpm
Liquid	SG Blowdown Treatment	DRS-3200/4200	—	—	—	1.2E+05 cpm

3 Area Rad Levels

Alert — RA3 Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown

1 2 3 4 5 6 D

RA3.1
Dose rates > 15 mR/hr in **any** of the following areas:
- Unit 1 Control Room (ERS-7401)
- Unit 2 Control Room (ERS-8401)
- Central Alarm Station (by survey)
- Secondary Alarm Station (by survey)

RA3.2
An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 rooms or areas (Note 5)

Table R-2 Safe Operation & Shutdown Areas

Room / Area	Mode(s)
Auxiliary Building 573'	4, 5
Auxiliary Building 587' (including D/G room)	1, 2, 3, 4, 5
Auxiliary Building 591'	1, 2, 3, 4
Auxiliary Building 609' (including 4kV room)	1, 2, 3, 4, 5
Auxiliary Building 633'	1, 2, 3, 4
Turbine Building (**All** levels)	1, 2, 3
Turbine Building 591'	4, 5
Screenhouse	1, 2, 3, 4, 5

H — Hazards

1 Security

General Emergency — None

Site Area Emergency — HS1 HOSTILE ACTION within the plant PROTECTED AREA

1 2 3 4 5 6 D

HS1.1
A HOSTILE ACTION is occurring or has occurred within the plant PROTECTED AREA as reported by the Security Shift Supervisor

Alert — HA1 HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

1 2 3 4 5 6 D

HA1.1
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor
OR
A validated notification from NRC of an aircraft attack threat within 30 min. of the site

Unusual Event — HU1 Confirmed SECURITY CONDITION or threat

1 2 3 4 5 6 D

HU1.1
A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervisor
OR
Notification of a credible security threat directed at the site
OR
A validated notification from the NRC providing information of an aircraft threat

2 Seismic Event

General Emergency — None

Site Area Emergency — None

Alert — None

Unusual Event — HU2 Seismic event greater than OBE levels

1 2 3 4 5 6 D

HU2.1
Control Room personnel feel an actual or potential seismic event
AND
The occurrence of a seismic event is confirmed in manner deemed appropriate by the Shift Manager

3 Natural or Technical Hazard

General Emergency — None

Site Area Emergency — None

Alert — None

Unusual Event — HU3 Natural or Technological Hazard

1 2 3 4 5 6 D

HU3.1
A tornado strike within the PROTECTED AREA

HU3.2
Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode

HU3.3
Movement of personnel within the plant PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

HU3.4
A hazardous event that results in onsite conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)

4 Fire

General Emergency — None

Site Area Emergency — None

Table H-1 Fire Areas
- Control Room
- Containment
- Auxiliary Building
- Switchgear Areas
- Diesel Generator Rooms
- ESW System Enclosures
- AFW Pump Rooms
- Refueling Water Storage Tank
- Condensate Storage Tank

Unusual Event — HU4 FIRE potentially degrading the level of safety of the plant

1 2 3 4 5 6 D

HU4.1
A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):
- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND
The FIRE is located within **any** Table H-1 area

HU4.2
Receipt of a single fire alarm (i.e., **no** other indications of a FIRE)
AND
The fire alarm is indicating a FIRE within **any** Table H-1 area
AND
The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1)

HU4.3
A FIRE within the plant or ISFSI PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)

HU4.4
A FIRE within the plant or ISFSI PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

5 Hazardous Gases

General Emergency — None

Table H-2 Safe Operation & Shutdown Areas

Room / Area	Mode(s)
Auxiliary Building 573'	4, 5
Auxiliary Building 587' (including D/G room)	1, 2, 3, 4, 5
Auxiliary Building 591'	1, 2, 3, 4
Auxiliary Building 609' (including 4kV room)	1, 2, 3, 4, 5
Auxiliary Building 633'	1, 2, 3, 4
Turbine Building (**All** levels)	1, 2, 3
Turbine Building 591'	4, 5
Screenhouse	1, 2, 3, 4, 5

Alert — HA5 Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

1 2 3 4 5 6 D

HA5.1
Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 rooms or areas
AND
Entry into the room or area is prohibited or IMPEDED (Note 5)

Unusual Event — None

6 Control Room Evacuation

General Emergency — None

Site Area Emergency — HS6 Inability to control a key safety function from outside the Control Room

1 2 3 4 5 6

HS6.1
An event has resulted in plant control being transferred from the Control Room to the Local Shutdown Instrumentation
AND
Control of **any** of the following key safety functions is **not** reestablished within 15 min. (Note 1):
- Reactivity control (modes 1, 2 and 3 **only**)
- Core cooling
- RCS heat removal

Alert — HA6 Control Room evacuation resulting in transfer of plant control to alternate locations

1 2 3 4 5 6 D

HA6.1
An event has resulted in plant control being transferred from the Control Room to the Local Shutdown Instrumentation

Unusual Event — None

7 SEC Judgment

General Emergency — HG7 Other conditions exist which in the judgment of the Site Emergency Coordinator warrant declaration of a General Emergency

1 2 3 4 5 6 D

HG7.1
Other conditions exist which in the judgment of the SEC indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area

Site Area Emergency — HS7 Other conditions existing that in the judgment of the Site Emergency Coordinator warrant declaration of a Site Area Emergency

1 2 3 4 5 6 D

HS7.1
Other conditions exist which in the judgment of the SEC indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Alert — HA7 Other conditions exist that in the judgment of the Site Emergency Coordinator warrant declaration of an Alert

1 2 3 4 5 6 D

HA7.1
Other conditions exist which, in the judgment of the SEC, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Unusual Event — HU7 Other conditions existing that in the judgment of the Site Emergency Coordinator warrant declaration of a UE

1 2 3 4 5 6 D

HU7.1
Other conditions exist which in the judgment of the SEC 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.

E — ISFSI

General Emergency — None

Site Area Emergency — None

Alert — None

Unusual Event — EU1 Damage to a loaded cask CONFINEMENT BOUNDARY

1 2 3 4 5 6 D

EU1.1
Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading:
- > 60 mrem/hr (gamma + neutron) on the top of the overpack
- > 600 mrem/hr (gamma + neutron) on the side of the overpack excluding inlet and outlet ducts

Modes:

1	2	3	4	5	6	D
Power Operation	Startup	Hot Standby	Hot Shutdown	Cold Shutdown	Refueling	Defueled

AEP AMERICAN ELECTRIC POWER — AEP: America's Energy Partner

AEP: D.C. Cook
Classification of Emergency
PMP-2080-EPP-101 Draft H4

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

S — System Malfunct.

1 Loss of Emerg AC Power

General Emergency — SG1 Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses

1 2 3 4

SG1.1
Loss of **all** offsite and **all** onsite AC power to emergency 4.16KV buses T11A (T21A) and T11D (T21D)
AND EITHER:
- Restoration of at least one essential bus in < 4 hours is **not** likely (Note 1)
- CSFST Core Cooling RED PATH (F-0.2) conditions met

Site Area Emergency — SS1 Loss of **all** offsite and **all** onsite AC power to emergency buses for 15 minutes or longer

1 2 3 4

SS1.1
Loss of **all** offsite and **all** onsite AC power to emergency 4.16KV buses T11A (T21A) and T11D (T21D) for ≥ 15 min. (Note 1)

Alert — SA1 Loss of **all** but one AC power source to emergency buses for 15 minutes or longer

1 2 3 4

SA1.1
AC power capability, Table S-1, to emergency 4.16KV buses T11A (T21A) and T11D (T21D) reduced to a single power source for ≥ 15 min. (Note 1)
AND
Any additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS

Unusual Event — SU1 Loss of **all** offsite AC power capability to emergency buses for 15 minutes or longer

1 2 3 4

SU1.1
Loss of **all** offsite AC power capability, Table S-1, to emergency 4.16KV buses T11A (T21A) and T11D (T21D) for ≥ 15 min. (Note 1)

Table S-1 AC Power Sources

Offsite:
- Unit Auxiliary Xmr TR1AB (TR2AB)
- Unit Auxiliary Xmr TR1CD (TR2CD)
- Reserve Auxiliary Xmr TR101AB (TR201AB)
- Reserve Auxiliary Xmr TR101CD (TR201CD)
- 69/4.16 kV Alternate Xfr TR12EP-1

Onsite:
- EDG 1AB (2AB)
- EDG 1CD (2CD)

2 Loss of Vital DC Power

General Emergency — SG2 Loss of **all** emergency AC and vital DC power sources for 15 minutes or longer

1 2 3 4

SG2.1
Loss of **all** offsite and **all** onsite AC power to emergency 4.16KV buses T11A (T21A) and T11D (T21D) for ≥ 15 min.
AND
Loss of **all** 250 VDC power based on bus voltage indications < 215 VDC on **all** vital DC buses 1CD (2CD) (Train A) and 1AB (2AB) (Train B) for ≥ 15 min.
(Note 1)

Site Area Emergency — SS2 Loss of **all** vital DC power for 15 minutes or longer

1 2 3 4

SS2.1
Loss of **all** 250 VDC power based on bus voltage indications < 215 VDC on **all** vital DC buses 1CD (2CD) (Train A) and 1AB (2AB) (Train B) for ≥ 15 min. (Note 1)

Alert — None

3 Loss of CR Indications

General Emergency — None

Table S-3 Significant Transients
- Reactor trip
- Runback > 25% thermal power
- Electrical load rejection > 25% of full electrical load
- ECCS actuation

Alert — SA3 UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress

1 2 3 4

SA3.1
An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)
AND
Any significant transient is in progress, Table S-3

Unusual Event — SU3 UNPLANNED loss of Control Room indications for 15 minutes or longer

1 2 3 4

SU3.1
An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

4 RCS Activity

General Emergency — None

Site Area Emergency — None

Table S-2 Safety System Parameters
- Reactor power
- RCS level
- RCS pressure
- Core Exit TC temperature
- Level in at least one S/G
- Auxiliary feed flow in at least one SG

Unusual Event — SU4 RCS activity greater than Technical Specification allowable limits

1 2 3 4

SU4.1
Sample analysis indicates RCS activity > Technical Specification 3.4.16 limits

5 RCS Leakage

General Emergency — None

Site Area Emergency — None

Alert — None

Unusual Event — SU5 RCS leakage for 15 minutes or longer

1 2 3 4

SU5.1
RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min.
OR
RCS identified leakage > 25 gpm for ≥ 15 min. (Note 1)
OR
Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min.

6 RPS Failure

General Emergency — None

Site Area Emergency — SS6 Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

1

SS6.1
An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%
AND
All actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5%
AND EITHER:
- Core Cooling RED PATH (F-0.2) conditions met
- Heat Sink RED PATH (F-0.3) conditions met

Alert — SA6 Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control console are not successful in shutting down the reactor

1

SA6.1
An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%
AND
Manual trip actions taken at the reactor control console (reactor trip switches) are **not** successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8)

Unusual Event — SU6 Automatic or manual trip fails to shut down the reactor

1

SU6.1
An automatic trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** RPS setpoint is exceeded
AND
A subsequent automatic trip or manual trip action taken at the reactor control console (reactor trip switches) is successful in shutting down the reactor as indicated by reactor power< 5% (Note 8)

SU6.2
A manual trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** manual trip action was initiated
AND
A subsequent automatic trip or manual trip action taken at the reactor control console (reactor trip switches) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

7 Loss of Comm.

General Emergency — None

Site Area Emergency — None

Table S-4 Communication Methods

System	Onsite	ORO	NRC
Plant Page	X		
Plant Radios	X	X	
Plant Telephone	X	X	X
ENS Line		X	X
Commercial Telephone		X	X
Microwave Transmission		X	X

Unusual Event — SU7 Loss of **all** onsite or offsite communications capabilities

1 2 3 4

SU7.1
Loss of **all** Table S-4 onsite communication methods
OR
Loss of **all** Table S-4 ORO communication methods
OR
Loss of **all** Table S-4 NRC communication methods

8 CNMT Failure

General Emergency — None

Site Area Emergency — None

Alert — None

Unusual Event — SU8 Failure to isolate containment or loss of containment pressure control

1 2 3 4

SU8.1
Any penetration is **not** isolated within 15 min. of a VALID containment isolation signal
OR
Containment pressure > 2.8 psig with < one full train of containment depressurization equipment operating per design for ≥ 15 min. (Note 9)
(Note 1)

9 Hazardous Event Affecting Safety Systems

General Emergency — None

Table S-5 Hazardous Events
- Seismic event (earthquake)
- Internal or external FLOODING event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Other events with similar hazard characteristics as determined by the SEC

Alert — SA9 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

1 2 3 4

SA9.1
The occurrence of **any** Table S-5 hazardous event
AND EITHER:
- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Unusual Event — None

F — Fission Product Barriers

General Emergency — FG1.1 1 2 3 4
Loss of **any** two barriers
AND
Loss or potential loss of third barrier (Table F-1)

Site Area Emergency — FS1.1 1 2 3 4
Loss or potential loss of **any** two barriers (Table F-1)

Alert — FA1.1 1 2 3 4
Any loss or **any** potential loss of **EITHER** Fuel Clad **OR** RCS (Table F-1)

Table F-2 Containment Radiation - R/hr VRA-1310 (2310) / 1410 (2410)

Monitor	FC Loss	RCS Loss	CNMT Potential Loss
VRA-1310 (2310)	1,000	200	9,100
VRA-1410 (2410)	700	140	6,300

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad (FC) Barrier		Reactor Coolant System (RCS) Barrier		Containment (CNMT) Barrier	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A. RCS or SG Tube Leakage	None	None	1. An automatic or manual ECCS (SI) actuation required by **EITHER**: • UNISOLABLE RCS leakage • SG tube RUPTURE	1. Operation of a standby charging pump is required by **EITHER**: • UNISOLABLE RCS leakage • SG tube leakage 2. CSFST Integrity-RED PATH (F-0.4) conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B. Inadequate Heat Removal	1. CSFST Core Cooling-RED PATH (F-0.2) conditions met	1. CSFST Core Cooling-ORANGE PATH (F-0.2) conditions met 2. CSFST Heat Sink-RED PATH (F-0.3) conditions met **AND** Heat Sink is required	None	1. CSFST Heat Sink-RED PATH (F-0.3) conditions met **AND** Heat Sink is required	None	1. CSFST Core Cooling-RED PATH (F-0.2) conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C. CNMT Radiation / RCS Activity	1. Containment radiation > Table F-2 column "FC Loss" 2. Dose equivalent I-131 coolant activity > 300 μCi/gm	None	1. Containment radiation > Table F-2 column "RCS Loss"	None	None	1. Containment radiation > Table F-2 column "CNMT Potential Loss"
D. CNMT Integrity or Bypass	None	None	None	None	1. Containment isolation is required **AND EITHER** • Containment integrity has been lost based on SEC judgment • UNISOLABLE pathway from Containment to the environment exists 2. Indications of RCS leakage outside of Containment	1. CSFST Containment-RED PATH (F-0.5) conditions met 2. Containment hydrogen concentration ≥ 4% 3. Containment pressure > 2.8 psig with < one full train of depressurization equipment operating per design for ≥ 15 min. (Notes 1, 9)
E. SEC Judgment	1. **Any** condition in the opinion of the SEC that indicates loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the SEC that indicates potential loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the SEC that indicates loss of the RCS barrier	1. **Any** condition in the opinion of the SEC that indicates potential loss of the RCS barrier	1. **Any** condition in the opinion of the SEC that indicates loss of the Containment barrier	1. **Any** condition in the opinion of the SEC that indicates potential loss of the Containment barrier

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then **no** emergency classification is warranted.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is **not** required.

Note 7: This EAL does **not** apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 8: A manual trip action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Note 9: One Containment Spray System train and one Containment Air Recirculation Fan comprise one full train of depressurization equipment.

Note 10: Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss of decay heat removal.

EAL-1 MODES 1, 2, 3 & 4

EAL-2 MODES 5, 6 & Defueled

R – Abnorm. Rad Levels / Rad Effluent

1 – Rad Effluent

GENERAL EMERGENCY — RG1 Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE (Modes: 1 2 3 4 5 6 D)

RG1.1
Reading on **any** Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4)

RG1.2
Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the site boundary (Note 4)

RG1.3
Field survey results indicate **EITHER** of the following at or beyond the site boundary:
- Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation.

(Notes 1, 2)

SITE AREA EMERGENCY — RS1 Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE (Modes: 1 2 3 4 5 6 D)

RS1.1
Reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4)

RS1.2
Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the site boundary (Note 4)

RS1.3
Field survey results indicate **EITHER** of the following at or beyond the site boundary:
- Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation.

(Notes 1, 2)

ALERT — RA1 Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE (Modes: 1 2 3 4 5 6 D)

RA1.1
Reading on **any** Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4)

RA1.2
Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the site boundary (Note 4)

RA1.3
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the site boundary for 60 min. of exposure (Notes 1, 2)

RA1.4
Field survey results indicate **EITHER** of the following at or beyond the site boundary:
- Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation.

(Notes 1, 2)

UNUSUAL EVENT — RU1 Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer (Modes: 1 2 3 4 5 6 D)

RU1.1
Reading on **any** Table R-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3)

RU1.2
Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2)

2 – Irradiated Fuel Event

GENERAL EMERGENCY — RG2 Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer (Modes: 1 2 3 4 5 6 D)

RG2.1
Spent fuel pool level **cannot** be restored to at least 0 ft. on 1(2)-RLI-502-CRI Spent Fuel Pit Level Indication for ≥ 60 min. (Note 1)

SITE AREA EMERGENCY — RS2 Spent fuel pool level at the top of the fuel racks (Modes: 1 2 3 4 5 6 D)

RS2.1
Lowering of spent fuel pool level to 0 ft. on 1(2)-RLI-502-CRI Spent Fuel Pit Level Indication

ALERT — RA2 Significant lowering of water level above, or damage to, irradiated fuel (Modes: 1 2 3 4 5 6 D)

RA2.1
Uncovery of irradiated fuel in the REFUELING PATHWAY

RA2.2
Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by High alarm on any of the following radiation monitors:
- VRS-1101/1201, Unit 1 Upper Containment
- VRS-2101/2201, Unit 2 Upper Containment
- R-5 Spent Fuel Area
- VRS-5006 Spent Fuel Area

RA2.3
Lowering of spent fuel pool level to 9 ft. 6 in. on 1(2)-RLI-502-CRI Spent Fuel Pit Level Indication (8 ft. 10 in. on local ruler)

UNUSUAL EVENT — RU2 UNPLANNED loss of water level above irradiated fuel (Modes: 1 2 3 4 5 6 D)

RU2.1
UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication
AND
UNPLANNED rise in corresponding area radiation levels as indicated by **any** of the following radiation monitors:
- VRS-1101/1201, Unit 1 Upper Containment
- VRS-2101/2201, Unit 2 Upper Containment
- R-5 Spent Fuel Area
- VRS-5006 Spent Fuel Area

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Unit Vent Noble Gas	VRS-1500 (2500)	3.3E+00 µCi/cc	3.3E-01 µCi/cc	3.3E-02 µCi/cc	4.2E-03 µCi/cc
Gaseous	Gland Seal Leakoff	SRA-1800 (2800)	1.6E+02 µCi/cc	1.6E+01 µCi/cc	1.6E+00 µCi/cc	1.4E-01 µCi/cc
Gaseous	Steam Jet Air Ejector	SRA-1900 (2900)	1.5E+04 µCi/cc	1.5E+03 µCi/cc	1.5E+02 µCi/cc	1.3E+01 µCi/cc
Liquid	Radwaste Effluent	RRS-1001	—	—	—	4.6E+04 cpm
Liquid	SG Blowdown	R-19	—	—	—	1.7E+03 cpm
Liquid	SG Blowdown	DRS-3100/4100	—	—	—	1.2E+04 cpm
Liquid	SG Blowdown Treatment	R-24	—	—	—	2.9E+04 cpm
Liquid	SG Blowdown Treatment	DRS-3200/4200	—	—	—	1.2E+05 cpm

3 – Area Rad Levels

ALERT — RA3 Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown (Modes: 1 2 3 4 5 6 D)

RA3.1
Dose rates > 15 mR/hr in **any** of the following areas:
- Unit 1 Control Room (ERS-7401)
- Unit 2 Control Room (ERS-8401)
- Central Alarm Station (by survey)
- Secondary Alarm Station (by survey)

RA3.2
An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 rooms or areas (Note 5)

Table R-2 Safe Operation & Shutdown Areas

Room / Area	Mode(s)
Auxiliary Building 573'	4, 5
Auxiliary Building 587' (including D/G room)	1, 2, 3, 4, 5
Auxiliary Building 591'	1, 2, 3, 4
Auxiliary Building 609' (including 4kV room)	1, 2, 3, 4, 5
Auxiliary Building 633'	1, 2, 3, 4
Turbine Building (**All** levels)	1, 2, 3
Turbine Building 591'	4, 5
Screenhouse	1, 2, 3, 4, 5

H – Hazards

1 – Security

GENERAL EMERGENCY — None

SITE AREA EMERGENCY — HS1 HOSTILE ACTION within the plant PROTECTED AREA (Modes: 1 2 3 4 5 6 D)

HS1.1
A HOSTILE ACTION is occurring or has occurred within the plant PROTECTED AREA as reported by the Security Shift Supervisor

ALERT — HA1 Hostile action within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes (Modes: 1 2 3 4 5 6 D)

HA1.1
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor
OR
A validated notification from NRC of an aircraft attack threat within 30 min. of the site

UNUSUAL EVENT — HU1 Confirmed SECURITY CONDITION or threat (Modes: 1 2 3 4 5 6 D)

HU1.1
A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervisor
OR
Notification of a credible security threat directed at the site
OR
A validated notification from the NRC providing information of an aircraft threat

2 – Seismic Event

GENERAL EMERGENCY — None | **SITE AREA EMERGENCY** — None | **ALERT** — None

UNUSUAL EVENT — HU2 Seismic event greater than OBE levels (Modes: 1 2 3 4 5 6 D)

HU2.1
Control Room personnel feel an actual or potential seismic event
AND
The occurrence of a seismic event is confirmed in manner deemed appropriate by the Shift Manager

3 – Natural or Technical Hazard

GENERAL EMERGENCY — None | **SITE AREA EMERGENCY** — None | **ALERT** — None

UNUSUAL EVENT — HU3 Natural or Technological Hazard (Modes: 1 2 3 4 5 6 D)

HU3.1
A tornado strike within the PROTECTED AREA

HU3.2
Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode

HU3.3
Movement of personnel within the plant PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

HU3.4
A hazardous event that results in onsite conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)

4 – Fire

GENERAL EMERGENCY — None | **SITE AREA EMERGENCY** — None | **ALERT** — None

Table H-1 Fire Areas
- Control Room
- Containment
- Auxiliary Building
- Switchgear Areas
- Diesel Generator Rooms
- ESW System Enclosures
- AFW Pump Rooms
- Refueling Water Storage Tank
- Condensate Storage Tank

UNUSUAL EVENT — HU4 FIRE potentially degrading the level of safety of the plant (Modes: 1 2 3 4 5 6 D)

HU4.1
A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1)
- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND
The FIRE is located within **any** Table H-1 area

HU4.2
Receipt of a single fire alarm (i.e., **no** other indications of a FIRE)
AND
The fire alarm is indicating a FIRE within **any** Table H-1 area
AND
The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1)

HU4.3
A FIRE within the plant or ISFSI PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)

HU4.4
A FIRE within the plant or ISFSI PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

5 – Hazardous Gases

GENERAL EMERGENCY — None | **SITE AREA EMERGENCY** — None

Table H-2 Safe Operation & Shutdown Areas

Room / Area	Mode(s)
Auxiliary Building 573'	4, 5
Auxiliary Building 587' (including D/G room)	1, 2, 3, 4, 5
Auxiliary Building 591'	1, 2, 3, 4
Auxiliary Building 609' (including 4kV room)	1, 2, 3, 4, 5
Auxiliary Building 633'	1, 2, 3, 4
Turbine Building (**All** levels)	1, 2, 3
Turbine Building 591'	4, 5
Screenhouse	1, 2, 3, 4, 5

ALERT — HA5 Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown (Modes: 1 2 3 4 5 6 D)

HA5.1
Release of a toxic, corrosive, asphyxiant or flammable gas into any Table H-2 rooms or areas
AND
Entry into the room or area is prohibited or IMPEDED (Note 5)

UNUSUAL EVENT — None

6 – Control Room Evacuation

GENERAL EMERGENCY — None

SITE AREA EMERGENCY — HS6 Inability to control a key safety function from outside the Control Room (Modes: 1 2 3 4 5 6)

HS6.1
An event has resulted in plant control being transferred from the Control Room to the Local Shutdown Instrumentation
AND
Control of any of the following key safety functions is **not** reestablished within 15 min. (Note 1):
- Reactivity control (modes 1, 2 and 3 **only**)
- Core cooling
- RCS heat removal

ALERT — HA6 Control Room evacuation resulting in transfer of plant control to alternate locations (Modes: 1 2 3 4 5 6 D)

HA6.1
An event has resulted in plant control being transferred from the Control Room to the Local Shutdown Instrumentation

UNUSUAL EVENT — None

7 – SEC Judgment

GENERAL EMERGENCY — HG7 Other conditions exist which in the judgment of the Site Emergency Coordinator warrant declaration of a GENERAL EMERGENCY (Modes: 1 2 3 4 5 6 D)

HG7.1
Other conditions exist which in the judgment of the SEC indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area

SITE AREA EMERGENCY — HS7 Other conditions existing that in the judgment of the Site Emergency Coordinator warrant declaration of a SITE AREA EMERGENCY (Modes: 1 2 3 4 5 6 D)

HS7.1
Other conditions exist which in the judgment of the SEC indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

ALERT — HA7 Other conditions exist that in the judgment of the Site Emergency Coordinator warrant declaration of an ALERT (Modes: 1 2 3 4 5 6 D)

HA7.1
Other conditions exist which, in the judgment of the SEC, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

UNUSUAL EVENT — HU7 Other conditions existing that in the judgment of the Site Emergency Coordinator warrant declaration of a UE (Modes: 1 2 3 4 5 6 D)

HU7.1
Other conditions exist which in the judgment of the SEC 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.

E – ISFSI

GENERAL EMERGENCY — None | **SITE AREA EMERGENCY** — None | **ALERT** — None

UNUSUAL EVENT — EU1 Damage to a loaded cask CONFINEMENT BOUNDARY (Modes: 1 2 3 4 5 6 D)

EU1.1
Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading:
- > 60 mrem/hr (gamma + neutron) on the top of the overpack
- > 600 mrem/hr (gamma + neutron) on the side of the overpack excluding inlet and outlet ducts

C – Cold SD/ Refuel System Malfunct.

1 – RCS Level

GENERAL EMERGENCY — CG1 Loss of RCS inventory affecting fuel clad integrity with containment challenged (Modes: 5 6)

CG1.1
RCS level **cannot** be monitored for ≥ 30 min. (Note 1)
AND
Core uncovery is indicated by **any** of the following:
- UNPLANNED increase in **any** Table C-1 sump/tank level of sufficient magnitude to indicate core uncovery
- High alarm on Containment radiation monitor VRA-1310 (2310) or VRA-1410(2410)
- Erratic Source Range Monitor indication

AND
Any Containment Challenge indication, Table C-2

SITE AREA EMERGENCY — CS1 Loss of RCS inventory affecting core decay heat removal capability (Modes: 5 6)

CS1.1
RCS level cannot be monitored for ≥ 30 min. (Note 1)
AND
Core uncovery is indicated by **any** of the following:
- UNPLANNED increase in **any** Table C-1 sump/tank level of sufficient magnitude to indicate core uncovery
- High alarm on Containment radiation monitor VRA-1310 (2310) or VRA-1410(2410)
- Erratic Source Range Monitor indication

ALERT — CA1 Significant loss of RCS inventory (Modes: 5 6)

CA1.1
Loss of RCS inventory as indicated by RCS level < 614.0 ft.

CA1.2
RCS water level cannot be monitored for ≥ 15 min. (Note 1)
AND EITHER
- UNPLANNED increase in **any** Table C-1 sump/tank level due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

UNUSUAL EVENT — CU1 UNPLANNED loss of RCS inventory (Modes: 5 6)

CU1.1
UNPLANNED loss of reactor coolant results in RCS water level less than a required lower limit for ≥ 15 min. (Note 1)

CU1.2
RCS water level cannot be monitored
AND EITHER
- UNPLANNED increase in **any** Table C-1 sump/tank level due to loss of RCS inventory
- Visual observation of unisolable RCS leakage

2 – Loss of Emerg AC Power

GENERAL EMERGENCY — None | **SITE AREA EMERGENCY** — None

ALERT — CA2 Loss of **all** offsite and **all** onsite AC power to emergency buses for 15 minutes or longer (Modes: 5 6 D)

CA2.1
Loss of **all** offsite and **all** onsite AC power to emergency 4.16KV buses T11A (T21A) and T11D (T21D) for ≥ 15 min. (Note 1)

UNUSUAL EVENT — CU2 Loss of all but one AC power source to emergency buses for 15 minutes or longer (Modes: 5 6 D)

CU2.1
AC power capability, Table C-3, to emergency 4.16 kV buses T11A (T21A) and T11D (T21D) reduced to a single power source for ≥ 15 min. (Note 1)
AND
Any additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS

3 – RCS Temp.

GENERAL EMERGENCY — None | **SITE AREA EMERGENCY** — None

ALERT — CA3 Inability to maintain plant in cold shutdown (Modes: 5 6)

CA3.1
UNPLANNED increase in RCS temperature to > 200°F for > Table C-4 duration (Note 1, 10)
OR
UNPLANNED RCS pressure increase > 10 psig (This EAL does not apply during water-solid plant conditions

UNUSUAL EVENT — CU3 UNPLANNED increase in RCS temperature (Modes: 5 6)

CU3.1
UNPLANNED increase in RCS temperature to > 200°F (Note 10)

CU3.2
Loss of **all** RCS temperature and RCS level indication for ≥ 15 min. (Note 1)

4 – Loss of Vital DC Power

GENERAL EMERGENCY — None | **SITE AREA EMERGENCY** — None | **ALERT** — None

UNUSUAL EVENT — CU4 Loss of Vital DC power for 15 minutes or longer (Modes: 5 6)

CU4.1
< 215 VDC bus voltage indications on Technical Specification **required** 250 VDC vital buses for ≥ 15 min. (Note 1)

5 – Loss of Comm.

GENERAL EMERGENCY — None | **SITE AREA EMERGENCY** — None | **ALERT** — None

UNUSUAL EVENT — CU5 Loss of **all** onsite or offsite communications capabilities (Modes: 5 6 D)

CU5.1
Loss of **all** Table C-5 onsite communication methods
OR
Loss of **all** Table C-5 ORO communication methods
OR
Loss of **all** Table C-5 NRC communication methods

6 – Hazardous Event Affecting Safety Systems

GENERAL EMERGENCY — None | **SITE AREA EMERGENCY** — None

ALERT — CA6 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode (Modes: 5 6)

CA6.1
The occurrence of **any** Table C-6 hazardous event
AND EITHER:
- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

UNUSUAL EVENT — None

Table C-1 Sumps / Tanks
- Containment Sumps
- Auxiliary Building Sumps
- RWST
- RCDT

Table C-2 Containment Challenge Indications
- CONTAINMENT CLOSURE **not** established (Note 6)
- Containment hydrogen concentration ≥ 4%
- UNPLANNED rise in Containment pressure

Table C-3 AC Power Sources

Offsite:
- Reserve Auxiliary Xmr TR101AB (TR201AB)
- Reserve Auxiliary Xmr TR101CD (TR201CD)
- 69/4.16 kV Alternate Xmr TR12EP-1
- Main Xmr TR1 (TR2) backfeed (only if already aligned)

Onsite:
- EDG 1AB (2AB)
- EDG 1CD (2CD)

Table C-4 RCS Reheat Duration Thresholds

RCS Status	Containment Closure Status	Heat-up Duration
INTACT (but **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

Table C-5 Communication Methods

System	Onsite	ORO	NRC
Plant Page	X		
Plant Radios	X	X	
Plant Telephone	X	X	X
ENS Line		X	X
Commercial Telephone		X	X
Microwave Transmission		X	X

Table C-6 Hazardous Events
- Seismic event (earthquake)
- Internal or external FLOODING event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Other events with similar hazard characteristics as determined by the SEC

Notes

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a GENERAL EMERGENCY is not required.

Note 7: This EAL does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

Note 9: One Containment Spray System train and one Containment Air Recirculation Fan comprise one full train of depressurization equipment.

Note 10: Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss of decay heat removal.

Modes: 1 Power Operation | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refueling | D Defueled

Enclosure 7 to AEP-NRC-2017-02

DONALD C. COOK NUCLEAR PLANT EMERGENCY PLAN
EAL TECHNICAL BASES CALCULATIONS

EP-CALC-CNP-1601, Radiological Effluent EAL Threshold Values

EP-CALC-CNP-1602, Containment Radiation EAL Threshold Values

1-2-UNC-421 Calc1, Post Accident High Range Containment Area Radiation Monitoring Loop Uncertainty Calculation

EVAL-RD-99-11 (Excerpts for EP-CALC-CNP-1602), Evaluation of Radiation Monitoring System Setpoints