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October 25, 2016

PG&E Letter DCL-16-099

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

10 CFR 50.90

Diablo Canyon Units 1 and 2
Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
License Amendment Request 16-04
Request to Adopt Emergency Action Level Schemes Pursuant to NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors"

References:
1. NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," dated November 2012 (ADAMS Accession No. ML12326A805)
2. Letter from Mark Thaggard (U.S. Nuclear Regulatory Commission) to Susan Perkins-Grew (Nuclear Energy Institute), "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)
3. NRC Regulatory Issue Summary (RIS) 2005-02, Revision 1, "Clarifying the Process for Making Emergency Plan Changes," dated April 19, 2011 (ADAMS Accession No. ML100340545)
4. NRC Order EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" (ADAMS Accession No. ML12056A044)

Dear Commissioners and Staff:

Pursuant to 10 CFR 50.90, Pacific Gas and Electric Company (PG&E) hereby requests approval of the enclosed proposed amendment to Facility Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively. The enclosed license amendment request (LAR) proposes to revise the Emergency Plan (E-Plan) for DCPP to adopt the Nuclear Energy Institute's (NEI's) revised Emergency Action Level (EAL) schemes described in NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-

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Passive Reactors," which have been endorsed by the NRC as documented in an NRC letter dated March 28, 2013 (Reference 2). The E-Plan, as changed, would continue to meet the standards in 10 CFR 50.47 and the requirements in Appendix E to 10 CFR 50.

DCPP's currently approved E-Plan EAL schemes are based on the guidance established in NEI 99-01, Revision 4, dated January 2003, except for the security-related EALs, which are from the guidance established in NEI 99-01, Revision 5, dated February 2008.

Federal Regulation 10 CFR 50, Appendix E, Section IV.B.2 requires that "a licensee desiring to change its entire emergency action level scheme to submit an application for an amendment to its license and receive NRC approval before implementing the change."

Regulatory Issue Summary (RIS) 2005-02, Revision 1 (Reference 3), provides guidance that a revision to an entire EAL scheme must be submitted for prior NRC approval as specified in Section IV.B of Appendix E to 10 CFR 50.

Therefore, pursuant to 10 CFR 50.90, PG&E hereby requests NRC review and approval of revisions to DCPP's E-Plan EALs. PG&E, thereby, proposes to adopt the EAL scheme based on the latest NRC-endorsed guidance as described in NEI 99-01, Revision 6. This revision to the EAL schemes also incorporates spent fuel pool emergency action levels required by NRC Order EA-12-051 (Reference 4).

The changes in this LAR are not required to address an immediate safety concern. PG&E requests approval of this LAR no later than October 19, 2017. PG&E requests the license amendments be made effective upon NRC issuance, to be implemented within 180 days from the NRC approval of the license amendment to permit program changes and training.

The enclosure to this letter contains the evaluation of the proposed change along with the following attachments:

Attachment 1: EAL Comparison Matrix
Attachment 2: EAL Technical Basis Document Markup
Attachment 3: EAL Technical Basis Document, Revised

PG&E makes no new or revised regulatory commitments (as defined by NEI 99-04) in this letter.

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92, "Issuance of amendment." Pursuant to 10 CFR 51.22, "Criterion for categorical exclusion;

identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," section (b), no environmental impact statement or environmental assessment need to be prepared in connection with the issuance of this amendment.

In accordance with site administrative procedures and the Quality Assurance Program, the proposed amendment has been reviewed by the Plant Staff Review Committee.

Pursuant to 10 CFR 50.91, "Notice for public comment; State consultation," PG&E is sending a copy of this proposed amendment to the California Department of Public Health.

If you have any questions or require additional information, please contact Mr. Hossein Hamzehee at 805-545-4720.

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

Executed on October 25, 2016.

Sincerely,

Paula Gerfen

Paula Gerfen
Station Director

e1d7/4418/50521100
Enclosure
cc: Diablo Distribution
cc/enc: Kriss Kennedy, NRC Region IV Administrator
Chris W. Newport, NRC Senior Resident Inspector
Gonzalo L. Perez, Branch Chief, California Department of Public Health
Balwant K. Singal, NRC Senior Project Manager

Evaluation of the Proposed Change

License Amendment Request 16-04
Request to Adopt Emergency Action Level Schemes Pursuant to NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors"

ATTACHMENTS:

1. EAL Comparison Matrix
2. EAL Technical Basis Document Markup
3. EAL Technical Basis Document, Revised

EVALUATION

1. SUMMARY DESCRIPTION

This letter is a request to amend Operating Licenses DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively.

The proposed change would revise the facility's currently approved Emergency Plan Emergency Action Level (EAL) scheme which are currently based on the guidance established in Nuclear Energy Institute (NEI) 99-01, Revision 4, dated January 2003 (Reference 1) and NEI 99-01, Revision 5, dated February 2008 (Reference 2) for security-related EALs, to the guidance established in NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," (Reference 3) which has been endorsed by the NRC (Reference 4).

The proposed changes to the EAL scheme contained in this submittal do not reduce the capability to meet the applicable emergency planning requirements established in 10 CFR 50.47, "Emergency Plans," and 10 CFR 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities."

Adopting NEI 99-01, Revision 6, will continue to provide consistent emergency classifications internally and externally. Federal Regulation 10 CFR 50, Appendix E, Section IV.B.2 requires prior NRC approval when a licensee is changing from one NRC-approved EAL scheme to another EAL scheme.

2. DETAILED DESCRIPTION

The proposed EAL changes were reviewed considering the requirements of 10 CFR 50.54(q), "Conditions of Licenses - Emergency Plans," paragraph (b) of 10 CFR 50.47, and 10 CFR 50 Appendix E, Regulatory Issue Summary (RIS) 2003-18, "Use of 99-01, Methodology for Development of Emergency Action Levels," (including supporting supplements, Reference 5), and RIS 2005-02, Revision 1, "Clarifying the Process for Making Emergency Plan Changes" (Reference 6). Adopting the proposed changes to the DCPP EAL scheme does not reduce the effectiveness of the Emergency Plan (E-Plan) and the Emergency Plan continues to comply with the standards established in 10 CFR 50.47 and 10 CFR 50, Appendix E.

The attached marked-up and clean copies of the EAL Technical Bases Document (TBD) (Attachments 2 and 3, respectively) provide an explanation and rationale for each EAL included in the EAL change. The EAL Basis document includes the necessary plant information.

The EAL Comparison Matrix in Attachment 1 provides a line-by-line comparison between the proposed DCPP Initiating Conditions and Mode Applicability and EAL wording with the Initiating Conditions and Mode Applicability, and the

NEI 99-01, Revision 6, example EAL wording. This document provides a means of assessing DCPP differences and deviations from the NRC-endorsed guidance given in NEI 99-01, Revision 6.

Discussion of each DCPP EAL bases and a list of source document references are given in the EAL TBD. It is therefore advisable to reference the EAL TBD for background information while using the Comparison Matrix.

2.1 Background

EALs are the plant-specific indications, conditions, or instrument readings that are utilized to classify emergency conditions defined in the DCPP E-Plan. In 1992, the NRC endorsed NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels," (Reference 7) as an alternative to NUREG-0654 EAL guidance. NEI 99-01 (NUMARC/NESP-007) Revisions 4 and 5 were subsequently issued for industry implementation. Enhancements over earlier revisions included:

- Consolidating the system malfunction initiating conditions and example EALs which address conditions that may be postulated to occur during plant shutdown conditions.
- Initiating conditions and example EALs 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 was issued which incorporates resolutions to numerous implementation issues including the NRC EAL frequently-asked questions. Using Reference 3, DCPP, in coordination with other members of the STARS Alliance, conducted an EAL implementation upgrade project that produced the EALs discussed herein.

3. TECHNICAL EVALUATION

DCPP's currently approved E-Plan EAL schemes are based on the guidance established in NEI 99-01, Revision 4, dated January 2003, and NEI 99-01, Revision 5, dated February 2008 for security-related EALs. The proposed change revises the E-Plan EAL scheme to be based on NEI 99-01, Revision 6. Within Attachment 1, the basis for each difference or deviation between NEI 99-01, Revision 6, guidance and the DCPP TBD is provided. The differences do not alter the meaning or intent of the Initiating Conditions (ICs) or EALs. The deviations from the NEI 99-01, Revision 6, guidance are justified and have been deemed acceptable deviations by NRC, as indicated in NRC Emergency Preparedness Frequently Asked Question (EPFAQ) Nos. 2015-013

(ADAMS Accession No. ML16166A366) and 2015-014 (ADAMS Accession No. ML16166A240). These changes affect only the DCPP E-Plan and otherwise do not alter requirements of the Operating License or the Technical Specifications. These changes do not alter any of the assumptions used in the safety analyses, nor do they cause any safety system parameters to exceed their acceptance limits. Therefore, the proposed changes have no adverse effect on plant safety.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.47(b)(4) requires the emergency 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:

> 1. The means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within and outside the site boundary to protect health and safety. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite and offsite monitoring. By June 20, 2012, for nuclear power reactor licensees, these action levels must include hostile action that may adversely affect the nuclear power plant. The initial emergency action levels shall be discussed and agreed on by the applicant or licensee and state and local governmental authorities, and approved by the NRC. Thereafter, emergency action levels shall be reviewed with the State and local governmental authorities on an annual basis.

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 § 50.54(q) for all other emergency action level changes.

The NRC endorsement letter of NEI 99-01, Revision 6, 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."

This section of Appendix E requires licensees desiring to change its entire emergency action level scheme to submit an application for amendment to its license and receive prior NRC approval before implementing the change. Consequently, this request for NRC approval of the proposed EAL scheme change is as specified in 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," and 10 CFR 50.4, "Written Communications."

4.2 Precedent

This request is similar in nature to requests for Braidwood Station, Units 1 and 2 (Reference 8), Byron Station, Units 1 and 2, Callaway Plant, Unit 1 (Reference 9), Clinton Power Station, Unit 1 , Dresden Nuclear Power Station, Units 1, 2, and 3, LaSalle County Station, Units 1 and 2 , Limerick Generating Station, Units 1 and 2, Oyster Creek Nuclear Generating Station, Peach Bottom Atomic Power Station, Units 1, 2, and 3 , Quad Cities Nuclear Power Station, Units 1 and 2, South Texas Project, Units 1 and 2 (Reference 10), Three Mile Island Nuclear Station, Units 1 and 2, and V.C. Summer Nuclear Station, Unit 1 (Reference 11).

4.3 Significant Hazards Consideration

Pacific Gas and Electric (PG&E) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on 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 or consequences of an accident previously evaluated?

Response: No.

The proposed changes to the Diablo Canyon Power Plant (DCPP) emergency action levels (EALs) do not impact the physical function of plant structures, systems, or components (SSCs) or the manner in which SSCs perform their design function. 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 change does 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 accident from any accident previously evaluated?

Response: No.

The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed or removed) or a change in the method of plant operation. The proposed changes will not introduce failure modes that could result in a new accident, and the change does not alter assumptions made in the safety analysis. The proposed changes to the DCPP EALs are not initiators of any accidents.

Therefore, the proposed change does not create the possibility of a new or different accident from any accident 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 do not impact operation of the plant or its response to transients or accidents. The proposed 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 E-Plan will continue to activate an emergency response commensurate with the extent of degradation of plant safety.

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

Based on the above evaluation, PG&E concludes that the proposed change does not involve a 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 Conclusions

In conclusion, based on PG&E's analysis of the no significant hazards consideration 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 Commission'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. ENVIRONMENTAL CONSIDERATION

The proposed changes to the EALs maintain the environmental bounds of the current environmental assessment associated with DCPP. The proposed changes will not affect plant safety and will not have an adverse effect on the probability of an accident occurring.

PG&E has evaluated the proposed amendment and has determined that 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 effluents 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. REFERENCES

1. NEI 99-01, Revision 4 (NUMARC/NESP-007), "Methodology for Development of Emergency Action Levels," dated January 2003 (ADAMS Accession No. ML030230250)

2. NEI 99-01, Revision 5, "Methodology for Development of Emergency Action Levels," dated February 2008 (ADAMS Accession No. ML080450149)

3. NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," dated November 2012 (ADAMS Accession No. ML12326A805)

4. Letter from Mark Thaggard (U.S. Nuclear Regulatory Commission) to Susan Perkins-Grew (Nuclear Energy Institute), "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)

5. NRC Regulatory Issue Summary 2003-18, "Use of NEI 99-01, 'Methodology for Development of Emergency Action Levels,' Revision 4, Dated January 2003," dated October 8, 2003

6. NRC Regulatory Issue Summary 2005-02, Revision 1, "Clarifying the Process for Making Emergency Plan Changes," dated April 19, 2011 (ADAMS Accession No. ML100340545)

7. NUMARC/NESP-007, Revision 2, "Methodology for Development of Emergency Action Levels," dated January 1992 (ADAMS Accession No. ML041120174)

8. Exelon Generation Letter RS-14-115, RA-14-032, TMI-14-046, "License Amendment Request to Adopt Emergency Action Level Schemes Pursuant to NEI 99-01, Revision 6, 'Development of Emergency Action Levels for Non-Passive Reactors,'" dated May 30, 2014 (ADAMS Accession No. ML14164A054)

9. Ameren Missouri Letter ULNRC-06143, "License Amendment Request for Emergency Action Level (EAL) Upgrade Adopting NRC-Endorsed NEI 99-01, Revision 6," dated October 2, 2014 (ADAMS Accession No. ML14275A441)

10. South Texas Project Nuclear Operating Company Letter NOC-AE-14003087, "License Amendment Request for Revision to Unit 1 and Unit 2 Emergency Action Levels," dated May 15, 2014 (ADAMS Accession No. ML14164A305)

11. South Carolina Electric and Gas letter RC-14-0032, "License Amendment Request LAR-14-02392 Request for NRC Approval of Proposed Changes to Emergency Action Levels," dated April 7, 2014 (ADAMS Accession No. ML14122A156)

EAL Comparison Matrix

Diablo Canyon Power Plant
NEI 99-01 Revision 6
EAL Comparison Matrix

10/12/16

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 Revision (Rev.) 6, "Development of Emergency Action Levels for Non-Passive Reactors," (ADAMS Accession Number ML12326A805), and the Diablo Canyon Power Plant (DCPP) ICs, Mode Applicability and EALs. This document provides a means of assessing DCPP differences and deviations from the NRC endorsed guidance given in NEI 99-01 Rev. 6. Discussion of DCPP 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 Rev. 6 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 Revision 6. Generally, each row of the comparison matrix provides the following information:

- NEI EAL/IC identifier
- NEI EAL/IC wording
- DCPP EAL/IC identifier
- DCPP 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."

EAL Emphasis Techniques

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

The print and paragraph formatting conventions summarized below guide presentation of the DCPP 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 Rev. 6.

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 Rev. 6 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 DCPP EALs appear as unique EALs (e.g., HU3.1 through HU3.4).
3. Mode applicability identifiers (numbers/letter) modify the NEI 99-01 Rev. 6 mode applicability names as follows: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown, 5 - Cold Shutdown, 6 - Refueling, D – Defueled. NEI 99-01 Rev. 6 defines 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 to reduce EAL-user reading burden, PG&E has adopted use of Boolean symbols in place of the NEI 99-01 Rev. 6 text modifiers within the EAL wording.
5. The term "Emergency Director" has been replaced by "Shift Manager/Site Emergency Coordinator/Emergency Director (SM/SEC/ED)" consistent with site-specific nomenclature.
6. Wherever the generic bracketed PWR term "reactor vessel/RCS" is provided, PG&E uses the term Reactor Coolant System or "RCS" as the site-specific nomenclature.
7. IC/EAL identification:
   - NEI Recognition Category A "Abnormal Radiation Levels/ Radiological Effluents" has been changed to Category R "Abnormal Rad Levels / Rad Effluents." The designator "R" is more intuitively associated with radiation (rad) or radiological events. NEI IC designators beginning with "A" have likewise been changed to "R."
   - NEI 99-01 Rev. 6 defines the thresholds requiring emergency classification (example EALs) and assigns them to ICs which, in turn, are grouped in "Recognition Categories." The DCPP 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 DCPP 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 DCPP 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 DCPP 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 DCPP 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 versus PWR).
- Using a numbering scheme other than that provided in NEI 99-01 Rev. 6 that does not change the intent of the overall scheme.
- Where the NEI 99-01 Rev. 6 guidance specifically provides an option to not include an EAL if equipment for the EAL does not exist at DCPP (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 versus at or above, less than or equal versus at or below, greater than or less than versus above or below, etc.)
- Adding DCPP equipment/instrument identification 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, EAL, or basis wording, as stated in NEI 99-01 Rev. 6, that does not alter the intent of the IC 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, and
    - Ensure that the resulting EAL scheme is complete (i.e., classifies all potential emergency conditions).

The following are examples of deviations:

- Use of altered mode applicability.
- Altering key words or time limits.
- Changing words of physical reference (protected area, safety-related equipment, etc.).
- Eliminating an IC. This includes the removal of an IC from the Fission Product Barrier Degradation category as this impacts the logic of Fission Product Barrier ICs.
- Changing a Fission Product Barrier from a Loss to a Potential Loss or vice-versa.
- Not using NEI 99-01 Rev. 6 definitions, as the intent is for all NEI 99-01 Rev. 6 users to have a standard set of defined terms as defined in NEI 99-01 Rev. 6. Differences due to plant types are permissible (BWR or PWR). Verbatim compliance to the wording in NEI 99-01 Rev. 6 is not necessary as long as the intent of the defined word is maintained. Use of the wording provided in NEI 99-01 Rev. 6 is encouraged since the intent is for all users to have a standard set of defined terms as defined in NEI 99-01 Rev. 6.
- Any change to the IC or EAL, or basis wording as stated in NEI 99-01 Rev. 6 that does alter the intent of the IC or EAL, i.e., the IC and/or EAL:
    - Does not classify at the classification level consistent with NEI 99-01 Rev. 6,
    - Is not logically integrated with other EALs in the EAL scheme, or
    - Results in an incomplete EAL scheme (i.e., does not classify all potential emergency conditions).

The "Difference Justification" columns in the remaining sections of this document identify each difference between the NEI 99-01 Rev. 6 IC/EAL wording and the DCPP 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 Rev. 6 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 Rev. 6.

PG&E has identified two deviations from the NEI 99-01 Rev. 6 guidance as represented in Table 3. These have been deemed by NRC to be acceptable deviations from the NEI 99-01 Rev. 6 guidance, as indicated in NRC Emergency Preparedness Frequently Asked Question (EPFAQ) Nos. 2015-013 (ADAMS Accession Number ML16166A366) and 2015-014 (ADAMS Accession Number ML16166A240).

Table 1 – DCPP EAL Categories/Subcategories

DCPP EALs		NEI
Category	**Subcategory**	**Recognition Category**
Group: Any Operating Mode:		
R – Abnormal Rad Levels/Rad Effluent	1 – Radiological Effluent 2 – Irradiated Fuel Event 3 – Area Radiation Levels	Abnormal **R**ad Levels/Radiological Effluent ICs/EALs
H – Hazards and Other Conditions Affecting Plant Safety	1 – Security 2 – Seismic Event 3 – Natural or Technological Hazard 4 – Fire 5 – Hazardous Gases 6 – Control Room Evacuation 7 – SM/SEC/ED 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 Essential AC Power 2 – Loss of Vital DC Power 3 – Loss of Control Room Indications 4 – RCS Activity 5 – RCS Leakage 6 – RTS 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 Vital 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 / DCPP EAL Identification Cross-Reference

NEI		DCPP	
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		DCPP	
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 Vital 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 Vital 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	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CS1.1

NEI		DCPP	
IC	Example EAL	Category and Subcategory	EAL
CS1	2	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CS1.2
CS1	3	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CS1.3
CG1	1	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CG1.1
CG1	2	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CG1.2
E-HU1	1	E – ISFSI	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	HU4.1
HU4	2	H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire	HU4.2
HU4	3	H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire	HU4.3

NEI		DCPP	
IC	Example EAL	Category and Subcategory	EAL
HU4	4	H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire	HU4.4
HU7	1	H – Hazards and Other Conditions Affecting Plant Safety, 7 – EC 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 – EC 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 – EC Judgment	HS7.1
HG1	1	N/A	N/A
HG7	1	H – Hazards and Other Conditions Affecting Plant Safety, 7 – EC Judgment	HG7.1
SU1	1	S – System Malfunction, 1 – Loss of Vital 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 – RTS Failure	SU6.1
SU5	2	S – System Malfunction, 6 – RTS Failure	SU6.2

NEI		DCPP	
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 Vital AC Power	SA1.1
SA2	1	S – System Malfunction, 3 – Loss of Control Room Indications	SA3.1
SA5	1	S – System Malfunction, 6 – RTS Failure	SA6.1
SA9	1	S – System Malfunction, 9 - Hazardous Event Affecting Safety Systems	SA9.1
SS1	1	S – System Malfunction, 1 – Loss of Vital AC Power	SS1.1
SS5	1	S – System Malfunction, 6 – RTS Failure	SS6.1
SS8	1	S – System Malfunction, 2 – Loss of Vital DC Power	SS2.1
SG1	1	S – System Malfunction, 1 – Loss of Vital AC Power	SG1.1
SG8	1	S – System Malfunction, 1 – Loss of Vital AC Power	SG2.1

Table 3 – Summary of Deviations

NEI		DCPP EAL	Description
IC	Example EAL		
HG1	1	N/A	Generic IC HG1 and associated example EAL are not implemented in the DCPP 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 Plant 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		DCPP 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 Rev. 6 and thus HG1 is adequately bounded as described above. **This is an acceptable deviation from the generic NEI 99-01 Rev. 6 guidance, as indicated in NRC EPFAQ No. 2015-013 (ADAMS Accession Number ML16166A366).**
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 Rev. 6 guidance, as indicated in NRC EPFAQ No. 2015-014 (ADAMS Accession Number ML16166A240).**

Category A

Abnormal Rad Levels / Radiological Effluent

NEI IC#	NEI IC Wording and Mode Applicability	DCPP IC#(s)	DCPP IC Wording and Mode Applicability	Difference 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 Offsite Dose Calculation Manual limits for 60 minutes or longer. MODE: All	The DCPP Offsite Dose Calculation Manual (ODCM) is the site-specific effluent release controlling document.

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 minutes. (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 DCPP monitored 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, conservatively 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 Offsite Dose Calculation Manual limits for ≥ 60 minutes. (Notes 1, 2)	The DCPP Offsite Dose Calculation Manual is the site-specific effluent release controlling document.

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 SM/SEC/ED 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 cannot be determined, 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording. The classification timeliness note has been standardized across the DCPP EAL scheme by referencing the "time limit" specified within the EAL wording. Changed the wording "is unknown" to "cannot be determined." Release start time would normally not be known until action is taken to determine when the release started. None

Table R-1 Effluent Monitor Classification Thresholds						
Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	Plant Vent	1(2)-RM-14/14R	-----	2.5E+6 cpm	2.5E+5 cpm	8.0E+4 cpm
				5.6E-2 μCi/cc	5.6E-3 μCi/cc	1.8E-3 μCi/cc
		1(2)-RM-87	1.8E-10 amps	----	----	----
			3.0E-1 μCi/cc			
Liquid	Liquid Radwaste Effluent Line	0-RM-18	-----	-----	-----	1.6E+5 cpm
	SGBD Tank	1(2)-RM-23	-----	-----	-----	2.0E+4 cpm

NEI IC#	NEI IC Wording and Mode Applicability	DCPP IC#(s)	DCPP IC Wording and Mode Applicability	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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 equivalent indication. **AND** UNPLANNED rise to low alarm setpoint in corresponding area radiation levels as indicated by **any** of the following radiation monitors: • RM-58 Spent Fuel Pool Area • RM-59 New Fuel Area • RM-2 Containment Area (Mode 6 **only**) • **Any** temporary installed monitor in vicinity of the REFUELING PATHWAY (when installed)	Added the word "…equivalent…" to indications to clarify that there may be multiple means of assessing a refueling cavity level drop. Added the words "…to low alarm setpoint…" to provide a low, but operationally significant, threshold for the operators to become aware of an area radiation level increase. The site-specific list of radiation monitors are listed in bullet format for ease of reading. RM-2 is applicable in Refueling Mode (6) only.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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 minutes. (Notes 1, 2, 3, 4)	The DCPP radiation monitors that detect radioactivity effluent release to the environment are listed in Table R-1. UE, Alert, SAE, and GE thresholds for all DCPP 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 minutes 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 are > 10 mR/hr and are expected to continue for ≥ 60 minutes. • Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 minutes of inhalation. (Notes 1, 2)	The site boundary is the site-specific receptor point. Added "…and are…" for clarification of intent.
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 SM/SEC/ED 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 cannot be determined, 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 **only** be used for emergency classification	The classification timeliness note has been standardized across the DCPP EAL scheme by referencing the "time limit" specified within the EAL wording. The classification timeliness note has been standardized across the DCPP EAL scheme by referencing the "time limit" specified within the EAL wording. Changed the wording "is unknown" to "cannot be determined." Release start time would normally not be known until action is taken to determine when the release started. None Incorporated site-specific EAL numbers associated with generic EAL#1. Added the word "**only**" to emphasize that the Table R-1

	assessments until the results from a dose assessment using actual meteorology are available.		assessments until the results from a dose assessment using actual meteorology are available.	effluent threshold values are only to be used in the absence of dose assessment using real meteorology being available.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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. **AND** High alarm on **any** of the following radiation monitors: • RM-59 New Fuel Storage Area • RM-58 Spent Fuel Pool Area • **Any** temporary installed monitor in vicinity of the REFUELING PATHWAY (when installed) • RM-2 Containment Area (Mode 6 **only**) • RM-44A/B Containment Ventilation Exhaust (Mode 6 **only**)	The NEI phrase "...from the fuel as indicated by ANY of the following radiation monitors" has been replaced with "...AND High alarm on **any** of the following radiation monitors:" for clarification that the classification requires two conditions: damage to fuel and a resultant high radiation alarm. 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 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 10 ft. above top of the fuel racks (Level 2).	Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 feet (ft.) above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3).

				SFP level instruments LI-801 and LI-801 provide SFP level indications in feet above top of the fuel racks. Level 1 = 23.75 ft., Level 2 = 10 ft., Level 3 = 1 ft. A project to add new Main Annunciator window PK11-04 alarm when Level 2 is reached is expected to be complete prior to PG&E implementation of NEI 99-01 Rev. 6.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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 (except RA3.2 – Modes 2, 3, 4 only)	EAL RA3.2 mode applicability is consistent with Table R-2 Area/Room assessment. Added the following note to the bases: *"**NOTE**: EAL RA3.2 mode applicability has been limited to the applicable modes identified in Table R-2 Safe Operation & Shutdown Rooms/Areas. If due to plant operating procedure or plant configuration changes, the applicable plant modes specified in Table R-2 are changed, a corresponding change to Attachment 3 'Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases' and to EAL RA3.2 mode applicability is required."*

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 **EITHER** of the following areas: Control Room (0-RM-1 or portable gamma radiation instrument) **OR** Central Alarm Station (by survey)	No other site-specific areas requiring continuous occupancy exist at DCPP. 0-RM-1 monitors the Control Room for area radiation. A portable gamma radiation instrument may be used when 0-RM-1 is out of service. The Central Alarm Station (CAS) does not have installed area radiation monitoring and thus must be determined by survey.
2	An UNPLANNED event results	RA3.2	An UNPLANNED event results in	Table R-2 contains the site-specific list of plant rooms or areas

	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)		radiation levels that prohibit or IMPEDE access to any Table R-2 rooms or areas. (Note 5)	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(s)
Auxiliary Building – 115' - BASTs	2, 3, 4
Auxiliary Building – 100' – BA Pumps	2, 3, 4
Auxiliary Building – 85' – Aux Control Board	2, 3, 4
Auxiliary Building – 64' – BART Tank area	2, 3, 4
Area H (below Control Room) – 100' 480V Bus area/rooms	3, 4

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

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 minutes. (Notes 1, 2, 3, 4)	The DCPP radiation monitors that detect radioactivity effluent release to the environment are listed in Table R-1. UE, Alert, SAE and GE thresholds for all DCPP 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 are > 100 mR/hr and are expected to continue for ≥ 60 minutes. • Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 minutes of	The site boundary is the site-specific receptor point. Added "… and are…" for clarification of intent.

	CDE greater than 500 mrem for one hour of inhalation.		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 SM/SEC/ED 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 cannot be determined, 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 **only** 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording. The classification timeliness note has been standardized across the DCPP EAL scheme by referencing the "time limit" specified within the EAL wording. Changed the wording "is unknown" to "cannot be determined." Release start time would normally not be known until action is taken to determine when the release started. None Incorporated site-specific EAL numbers associated with generic EAL#1. Added the word "**only**" to emphasize that the Table R-1 effluent threshold values are only to be used in the absence of dose assessment using real meteorology being available.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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. above top of the fuel racks (Level 3).	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). SFP level instruments LI-801 and LI-801 provide SFP level indications in feet above top of the fuel racks. Level 1 = 23.75 ft., Level 2 = 10 ft., Level 3 = 0 ft. (includes 1 ft. instrument uncertainly).

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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 minutes. (Notes 1, 2, 3, 4)	The DCPP radiation monitors that detect radioactivity effluent release to the environment are listed in Table R-1. UE, Alert, SAE, and GE thresholds for all DCPP 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	RG1.3	Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates are > 1,000 mR/hr and are expected to continue for ≥ 60 minutes. • Analyses of field survey samples indicate thyroid CDE > 5,000 mrem for 60 minutes of	The site boundary is the site-specific receptor point. Added "…and are…" for clarification of intent.

	CDE greater than 5,000 mrem for one hour of inhalation.		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 SM/SEC/ED 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 cannot be determined, 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 **only** 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording. The classification timeliness note has been standardized across the DCPP EAL scheme by referencing the "time limit" specified within the EAL wording. Changed the wording "is unknown" to "cannot be determined." Release start time would normally not be known until action is taken to determine when the release started. None Incorporated site-specific EAL numbers associated with generic EAL#1. Added the word "**only**" to emphasize that the Table R-1 effluent threshold values are only to be used in the absence of dose assessment using real meteorology being available.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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. above top of the fuel racks (Level 3) for ≥ 60 minutes. (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). SFP level instruments LI-801 and LI-801 provide SFP level indications in feet above top of the fuel racks. Level 1 = 23.75 ft., Level 2 = 10 ft., Level 3 = 1 ft. (includes 1 ft. instrument uncertainly).
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 SM/SEC/ED 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording.

Category C

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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." Since only one of the two EALs associated with IC CU1 has a timing component, the IC needs to reflect both EALs intent.

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 RCS inventory results in RCS water level less than a procedurally designated lower limit for ≥ 15 minutes. (Note 1)	Revised "reactor coolant" to "RCS inventory" to be consistent with the IC and CA1 wording. Replace "required" with "procedurally designated" to be specific as the source of the required level band.
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." Added bulleted criteria "Visual observation of UNISOLABLE RCS LEAKAGE" to include direct observation of RCS leakage. Table C-1 lists the site-specific sump and tanks that may be used as indirect RCS leakage indications based on level increases.
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 SM/SEC/ED should declare the event promptly upon determining that time limit has been	The classification timeliness note has been standardized across the DCPP 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 Structure Sumps • Reactor Cavity Sump • PRT • RCDT • CCW surge tank(s) • Auxiliary Building Sump • RWST • RHR Room Sumps (alarm **only**) • MEDT

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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 vital buses for 15 minutes or longer. MODE: 5 - Cold Shutdown, 6 - Refueling, D - Defueled	The DCPP vital buses are the site-specific emergency buses.

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 Unit 1 or Unit 2 vital 4.16KV buses F, G and H reduced to a single power source for ≥ 15 minutes. (Note 1) **AND** A failure of that single power source will result in loss of **all** AC power to SAFETY SYSTEMS.	4.16KV vital buses 1(2)F, 1(2)G and 1(2)H are the site-specific emergency buses. Site-specific AC power sources are listed in Table C-3. Reworded second conditional for clarity.
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 SM/SEC/ED 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording.

Table C-3 AC Power Capability		
	Unit 1	**Unit 2**
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

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

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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. (Note 10)	200 degrees Fahrenheit (°F) is the site-specific Technical Specification (Tech. Spec.) cold shutdown temperature limit. Added Note 10 to emphasize that hot condition EALs are subsequently applicable for any new event.
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 **all** RCS level indication for ≥ 15 minutes. (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 SM/SEC/ED 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording.
N/A	N/A	N/A	Note 10: Begin monitoring hot condition EALs concurrently.	Added Note 10 to emphasize that hot condition EALs are subsequently applicable for any new event.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference Justification
1	Indicated voltage is less than (site-specific bus voltage value) on required Vital DC buses for 15 minutes or longer.	CU4.1	< 105 VDC bus voltage indications on Technical Specification **required** 125 VDC buses for ≥ 15 minutes. (Note 1)	105 VDC is the site-specific minimum vital DC bus voltage. DC operability requirements are specified in Tech. Spec.
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 SM/SEC/ED 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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 offsite 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 of presentation. Table C-4 provides a site-specific list of onsite, offsite (ORO), and NRC communications methods.
2	Loss of **ALL** of the following ORO communications methods: (site specific list of communications methods)			
3	Loss of **ALL** of the following NRC communications methods: (site specific list of communications methods)			

Table C-5 Communication Methods			
System	**Onsite**	**Offsite**	**NRC**
Unit 1, Unit 2 and TSC Radio Consoles	X	X	
DCPP Telephone System (PBX)	X	X	X
Portable radio equipment (handie-talkies)	X		
Operations Radio System	X	X	
Security Radio Systems	X		
CAS and SAS Consoles	X	X	X
Fire Radio System	X		
Hot Shutdown Panel Radio Consoles	X	X	X
Public Address System	X		
NRC FTS			X
Mobile radios	X		
Satellite phones	X	X	X
Direct line (ATL) to the County and State OES		X	

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference Justification
CA1	Loss of (reactor vessel/RCS [*PWR*] or RCP [*BWR*]) inventory MODE: Cold Shutdown, Refueling	CA1	Loss of RCS inventory. MODE: 5 - Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 reactor vessel level < 107 ft. 6 in. (107.5 ft.) on RVRLIS, LI-400 standpipe or ultrasonic sensor. **OR** < 67.5% RVLIS full range (RVLIS equivalent to 107 ft. 6 in.).	When reactor vessel water level decreases to 107 ft. 6 inches (in.) elevation (el.), RCS level is approximately 21 in. above the bottom of the RCS hot leg penetration. This is the minimum procedurally allowed RCS level to preclude vortexing of the Residual Heat Removal (RHR) pumps while in Shutdown Cooling.
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 minutes. (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 bulleted criteria "Visual observation of UNISOLABLE RCS LEAKAGE" to include direct observation of RCS leakage. Table C-1 lists the site-specific sump and tanks that may be used as indirect RCS leakage indications based on level increases.
Note	The Emergency Director should declare the Alert promptly upon	N/A	Note 1: The SM/SEC/ED should declare the event promptly upon	The classification timeliness note has been standardized across the DCPP EAL scheme by referencing the "time limit" specified within the

	determining that 15 minutes has been exceeded, or will likely be exceeded		determining that time limit has been exceeded, or will likely be exceeded.	EAL wording.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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 vital buses for 15 minutes or longer. MODE: 5 -Cold Shutdown, 6 - Refueling, D - Defueled	The DCPP vital buses are the emergency buses.

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 capability, Table C-3, to Unit 1 or Unit 2 vital 4.16KV buses F, G and H for ≥ 15 minutes. (Note 1)	4.16KV buses 1(2)F, 1(2)G, and 1(2)H are the site-specific emergency buses. Site-specific AC power sources are tabularized 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 SM/SEC/ED 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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. (Note 1, 10) **OR** UNPLANNED RCS pressure increase > 10 psig (this 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. 10 psig is the site-specific pressure increase readable by Control Room indications. Added Note 10 to emphasize that hot condition EALs are subsequently applicable for any new event.
2	UNPLANNED RCS pressure increase greater than (site-specific pressure reading). (This EAL does not apply during water-solid plant conditions. [*PWR*])			
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SM/SEC/ED 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording.
N/A	N/A	N/A	Note 10: Begin monitoring hot condition EALs concurrently.	Added Note 10 to emphasize that hot condition EALs are subsequently applicable for any new event.

NEI:

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.		

DCPP:

Table C-4: RCS Heat-up Duration Thresholds		
RCS Status	**CONTAINMENT CLOSURE Status**	**Heat-up Duration**
INTACT (but **not** REDUCED INVENTORY)	N/A	60 minutes*
Not INTACT **OR** REDUCED INVENTORY	established	20 minutes*
	not established	0 minutes
* If an RCS heat removal system is in operation within this time frame and RCS temperature is trending down, the EAL is **not** applicable.		

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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 have been tabularized in Table C-6. Tsunami has been added as a site-specific hazard for DCPP.

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

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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: 5 - Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference Justification
1	a. CONTAINMENT CLOSURE not established. **AND** b. (Reactor vessel/RCS [*PWR*] or RCP [*BWR*]) level less than (site-specific level).	CS1.1	With CONTAINMENT CLOSURE **not** established, RVLIS full range < 62.1%. (Note 12)	When reactor vessel water level lowers to 62.1%, water level is six inches below the elevation of the bottom of the RCS hot leg penetration.
2	a. CONTAINMENT CLOSURE established. **AND** b. (Reactor vessel/RCS [*PWR*] or RCP [*BWR*]) level less than (site-specific level).	CS1.2	With CONTAINMENT CLOSURE established, RVLIS full range < 56.6% (Top of Fuel). (Note 12)	When reactor vessel water level drops below Reactor Vessel Level Instrumentation System (RVLIS) full range indication of 56.6% core uncovery is about to occur.
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:	CS1.3	RCS water level cannot be monitored for ≥ 30 minutes. (Note 1) **AND** Core uncovery is indicated by **any** of the following: • UNPLANNED increase in **any** Table C-1 sump/tank	Bridge (Manipulator) Crane Radiation Monitor > 9 Rad per hour (R/hr) (90% of instrument scale) would be indicative of possible core uncovery in the Refueling mode. Table C-1 lists the site-specific sump and tanks that may be used as indirect RCS leakage indications based on level increases.

	• (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)		level of sufficient magnitude to indicate core uncover. • **Any** Bridge (Manipulator) Crane Radiation Monitor > 9 R/hr. • 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 SM/SEC/ED 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording.
N/A	N/A	N/A	Note 12: With RVLIS out-of-service, classification shall be based on CS1.3 or CG1.2 if RCS inventory **cannot** be monitored.	Added Note 12 to emphasize that with RVLIS out of service, classification should be based on either CS1.3 or CG1.2 when RCS inventory cannot be monitored.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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: 5 - Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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).	CG1.1	RVLIS full range < 56.6% (Top of Fuel) for ≥ 30 minutes. (Note 1) **AND** **Any** Containment Challenge indication, Table C-2.	When reactor vessel water level drops below RVLIS full range indication of 56.6% core uncovery is about to occur. Table C-2 provides a tabularized list of containment challenge indications. 4% hydrogen concentration in the presence of oxygen represents an explosive mixture in containment.
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*]	CG1.2	RCS water level cannot be monitored for ≥ 30 minutes. (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 uncover. • **Any** Bridge (Manipulator) Crane Radiation Monitor	Bridge (Manipulator) Crane Radiation Monitor > 9 R/hr (90% of instrument scale) would be indicative of possible core uncovery in the Refueling mode. Table C-2 provides a tabularized list of containment challenge indications. 4% hydrogen concentration in the presence of oxygen represents an explosive mixture in containment.

	• 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).		> 9 R/hr. • Erratic Source Range 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 SM/SEC/ED 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 DCPP 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.
N/A	N/A	N/A	Note 12: With RVLIS out-of-service, classification shall be based on CS1.3 or CG1.2 if RCS inventory **cannot** be monitored.	Added Note 12 to emphasize that with RVLIS out of service, classification should be based on either CS1.3 or CG1.2 when RCS inventory cannot be monitored.

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	DCPP IC#(s)	DCPP IC Wording	Difference 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. DCPP is not a defueled station.

Category E

Independent Spent Fuel Storage Installation

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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 > Table E-1.	The DCPP ISFSI Technical Specifications do not have maximum contact dose rate specified for the exterior of an overpack. The values in Table E-1 are derived from the ISFSI UFSAR. Since the UFSAR Table 7.3-1A are the maximum calculated dose rate values, and are not expected to ever be exceeded, a conservative approach of exceeding the highest possible fuel value dose rates, plus 5 mRem/hour, was used as an indication of damage to an overpack. Note: These values are approximately 2 times the maximum expected dose rate for low burn-up fuel.

Table E-1 ISFSI Radiation Readings		
Dose Point Location (see figure)		**Surface Dose Rate** (mRem/hour)
1	Base vent	72
2	Mid plane	80
3	Top vent	76
4	Lid-center	22
4a	Lid-over top vents	139

Category F

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference Justification
1	Loss or Potential Loss of any two barriers.	FS1.1	Loss or potential loss of **any** two barriers. (Table F-1)	Table F-1 provides the fission product barrier loss and potential loss thresholds.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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.

Table F-1 Fission Product Barrier Threshold Matrix						
	Fuel Clad (FC) Barrier		**Reactor Coolant System (RCS) Barrier**		**Containment (CMT) 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 conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B **Inadequate Heat Removal**	1. CSFST Core Cooling-RED path conditions met	1. CSFST Core Cooling-MAGENTA path conditions met 2. CSFST Heat Sink-RED path conditions met **AND** Bleed and feed criteria met	None	1. CSFST Heat Sink-RED path conditions met **AND** Bleed and feed criteria met	None	1. CSFST Core Cooling-RED path conditions met **AND** Restoration procedures **not** effective within 15 minutes (Note 1)
C **CMT Radiation / RCS Activity**	1. Containment radiation (RM-30 or RM-31) > 300 R/hr 2. Dose equivalent I-131 coolant activity > 300 µCi/gm	None	1. Containment radiation (RM-30 or RM-31) > 40 R/hr	None	None	1. Containment radiation (RM-30 or RM-31) > 5,000 R/hr
D **CMT Integrity or Bypass**	None	None	None	None	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on SM/SEC/ED determination • UNISOLABLE pathway from Containment to the environment exists 2. Indications of RCS LEAKAGE outside of Containment	1. CSFST Containment-RED path conditions met (≥ 47 psig) 2. Containment hydrogen concentration ≥ 4% 3. Containment pressure ≥ 22 psig with < one full train of depressurization equipment operating per design for ≥ 15 minutes (Note 1, 9)
E **SM/SEC /ED Judgment**	1. **Any** condition in the opinion of the SM/SEC/ED that indicates loss of the fuel clad barrier	1. **Any** condition in the opinion of the SM/SEC/ED that indicates potential loss of the fuel clad barrier	1. **Any** condition in the opinion of the SM/SEC/ED that indicates loss of the RCS barrier	1. **Any** condition in the opinion of the SM/SEC/ED that indicates potential loss of the RCS barrier	1. **Any** condition in the opinion of the SM/SEC/ED that indicates loss of the containment barrier	1. **Any** condition in the opinion of the SM/SEC/ED that indicates potential loss of the containment barrier

PWR Fuel Clad Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI Threshold Wording	DCPP FPB #(s)	DCPP FPB Wording	Difference Justification
FC Loss 1	**RCS or SG Tube Leakage** Not Applicable	N/A	N/A	N/A
FC Loss 2	**Inadequate Heat Removal** A. Core exit thermocouple readings greater than (site-specific temperature value).	FC Loss B.1	CSFST Core Cooling-RED path conditions met.	Consistent with the generic developers note options Critical Safety Function Status Tree (CSFST) Core Cooling Red Path is used in lieu of Core Exit Thermocouples (CET) temperatures.
FC Loss 3	**RCS Activity/CMNT Rad** A. Containment radiation monitor reading greater than (site-specific value) **OR** B. (Site-specific indications that reactor coolant activity is greater than 300 μCi/gm dose equivalent I-131)	FC Loss C.1	Containment radiation (RM-30 or RM-31) > 300 R/hr.	Containment radiation monitor readings greater than 300 R/hr indicates 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.
		FC Loss C.2	Coolant activity > 300 μCi/gm Dose Equivalent I-131.	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 DCPP.

NEI FPB#	NEI Threshold Wording	DCPP FPB #(s)	DCPP FPB Wording	Difference Justification
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 SM/SEC/ED that indicates loss of the fuel clad barrier.	None
FC P-Loss 1	**RCS or SG Tube Leakage** A. RCS/reactor vessel level less than (site-specific level)	N/A	N/A	See FC P-Loss B.1. The RCS level threshold is implemented as CSFST Core Cooling Magenta 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-MAGENTA path conditions met.	Consistent with the generic developers note options CSFST Core Cooling MAGENTA Path is used in lieu of CET temperatures. MAGENTA is the DCPP specific path color equivalent of ORANGE path in the generic Pressurized Water Reactors Owners Group (PWROG) CSFSTs.
		FC P-Loss B.2	CSFST Heat Sink-RED path conditions met. **AND** Bleed and feed criteria met.	Consistent with the generic developers note options CSFST Heat Sink Red Path is used. For DCPP indication that heat removal is extremely challenged is manifested by CSFST Heat Sink RED path conditions met in combination with bleed and feed criteria being met. Refer to Attachment A for justification of incorporation of the bleed and feed condition threshold associated with loss of heat sink.
FC P-Loss 3	**RCS Activity/CMT Rad** Not Applicable	N/A	N/A	N/A
FC P-Loss 4	**CMT Integrity or Bypass** Not Applicable	N/A	N/A	N/A

NEI FPB#	NEI Threshold Wording	DCPP FPB #(s)	DCPP FPB Wording	Difference Justification
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 DCPP.
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 SM/SEC/ED that indicates potential loss of the fuel clad barrier.	None

PWR RCS Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI IC Wording	DCPP FPB #(s)	DCPP FPB Wording	Difference Justification
RCS Loss 1	**RCS or SG Tube Leakage** A. An automatic or manual ECCS (SI) actuation is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube RUPTURE.	RCS Loss A.1	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/CMNT Rad** A. Containment radiation monitor reading greater than (site-specific value).	RCS Loss C.1	Containment radiation (RM-30 or RM-31) > 40 R/hr.	Containment radiation monitor readings greater than 40 R/hr 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.
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 DCPP.

NEI FPB#	NEI IC Wording	DCPP FPB #(s)	DCPP FPB Wording	Difference Justification
RCS Loss 6	**Emergency Director Judgment** A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	RCS Loss E.1	**Any** condition in the opinion of the SM/SEC/ED that indicates loss of the RCS barrier.	None
RCS P-Loss 1	**RCS or SG Tube Leakage** A. Operation of a standby charging (makeup) pump is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube leakage. **OR** B. RCS cooldown rate greater than (site-specific pressurized thermal shock criteria/limits defined by site-specific indications).	RCS P-Loss A.1	Operation of a standby charging pump is required by **EITHER:** • UNISOLABLE RCS LEAKAGE. • SG tube leakage.	None
		RCS P-Loss A.2	CSFST Integrity-RED path 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 conditions met. **AND** Bleed and feed criteria met.	Consistent with the generic developers note options CSFST Heat Sink Red Path is used. For DCPP indication that heat removal is extremely challenged is manifested by CSFST Heat Sink RED path conditions met in combination with bleed and feed criteria being met. Refer to Attachment A for justification of incorporation of the bleed and feed condition threshold associated with loss of heat sink.

NEI FPB#	NEI IC Wording	DCPP FPB #(s)	DCPP FPB Wording	Difference Justification
RCS P-Loss 3	**RCS Activity/CMT Rad** Not Applicable	N/A	N/A	N/A
RCS P-Loss 4	**CMT 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 DCPP.
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 SM/SEC/ED that indicates potential loss of the RCS barrier.	None

PWR Containment Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI IC Wording	DCPP FPB #(s)	DCPP FPB Wording	Difference Justification
CMT Loss 1	**RCS or SG Tube Leakage** A. A leaking or RUPTURED SG is FAULTED outside of containment.	CMT Loss A.1	A leaking or RUPTURED SG is FAULTED outside of containment.	None
CMT Loss 2	**Inadequate Heat Removal** Not Applicable	N/A	N/A	N/A
CMT Loss 3	**RCS Activity/CMNT Rad** Not applicable	N/A	N/A	N/A
CMT Loss 4	**CMT 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.	CMT Loss D.1	Containment isolation is required. **AND EITHER:** • Containment integrity has been lost based on SM/SEC/ED determination. • UNISOLABLE pathway from containment to the environment exists.	Changed the word "judgment" to "determination" as containment integrity is a determinant condition which may include judgment.
		CMT Loss D.2	Indications of RCS LEAKAGE outside of containment.	None

NEI FPB#	NEI IC Wording	DCPP FPB #(s)	DCPP FPB Wording	Difference Justification
CMT Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific Containment Loss indication has been identified for DCPP.
CMT Loss 6	**Emergency Director Judgment** **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	CMT Loss E.1	**Any** condition in the opinion of the SM/SEC/ED that indicates loss of the containment barrier.	None
CMT P-Loss 1	**RCS or SG Tube Leakage** Not Applicable	N/A	N/A	N/A
CMT 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.	CMT P-Loss B.1	CSFST Core Cooling-RED path conditions met. **AND** Restoration procedures **not** effective within 15 minutes. (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.
CMT P-Loss 3	**RCS Activity/CMNT Rad** A. Containment radiation monitor reading greater than (site-specific value).	CMT P-Loss C.1	Containment radiation (RM-30 or RM-31) > 5,000 R/hr.	Containment radiation monitor readings greater than 5,000 R/hr indicate significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier.
CMT P-Loss 4	**CNMT Integrity or Bypass** A. Containment pressure greater than (site-specific value)	CMT P-Loss D.1	CSFST Containment-RED path conditions met (≥ 47 psig).	Consistent with the generic developers note options CSFST Containment Red Path is used in lieu of just containment pressure.

NEI FPB#	NEI IC Wording	DCPP FPB #(s)	DCPP FPB Wording	Difference 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.	CMT P-Loss D.2	Containment hydrogen concentration ≥ 4%.	4% hydrogen concentration in the presence of oxygen represents a flammable mixture in containment.
		CMT P-Loss D.3	Containment pressure ≥ 22 psig. **AND** Less than one full train of containment depressurization equipment operating per design for ≥ 15 minutes. (Note 1, 9)	The containment pressure setpoint (22 psig) is the pressure at which the equipment should actuate and begin performing its function. Added Note 1 consistent with other thresholds with a timing component. Added Note 9 to specify what constitutes a full train of containment depressurization equipment.
CMT 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 DCPP.
CMT P-Loss 6	**Emergency Director Judgment** A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.	CMT P-Loss E.1	**Any** condition in the opinion of the SM/SEC/ED that indicates potential loss of the containment barrier.	None

Category H

Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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 Watch Commander. **OR** Notification of a credible security threat directed at the site. **OR** A validated notification from the NRC providing information of an aircraft threat.	Example EALs #1, 2, and 3 have been combined into a single EAL for ease of presentation and use. The Security Watch Commander is defined as the 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	DCPP IC#(s)	DCPP IC Wording	Difference Justification
HU2	Seismic event greater than OBE level MODE: All	HU2	Seismic event greater than Design Earthquake (DE) level. MODE: All	An Operating Basis Earthquake (OBE) is referred to as Design Earthquake (DE) at DCPP, and a Safe Shutdown Earthquake (SSE) is referred to as Double Design Earthquake (DDE) at DCPP.

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference Justification
1	Seismic event greater than Operating Basis Earthquake (OBE) as indicated by: (site-specific indication that a seismic event met or exceeded OBE limits)	HU2.1	Seismic event > DE PGA as indicated by ground acceleration > 0.2 g on the "X" or "Y" axis or > 0.133 g on the "Z" axis. (Note 11)	If the EFM indicator alarms (> 0.2 g on the "X" or "Y" axis or > 0.133g on the "Z" axis) the DE has likely been exceeded.
N/A	N/A	N/A	Note 11: If the Earthquake Force Monitor (EFM) is out of service, refer to CP M-4 Earthquake for alternative methods to assess earthquakes.	Added Note 11 to provide guidance for assessing earthquakes if the EFM is out of service. A true determination of DE exceedance determination can take up to 4 hours, not 15 minutes (a more detailed explanation is in the Basis Background section). With this in mind, EAL declaration must be timely (within 15 minutes) and be based on the ground acceleration values from the EFM.

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

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference Justification
1	A tornado strike within the PROTECTED AREA.	HU3.1	A TORNADO strike within the PLANT PROTECTED AREA.	Added the word "PLANT" to protected area to distinguish from the ISFSI protected area that is located outside the plant protected area.
2	Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.	HU3.2	Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component required for the current operating mode. (Note 5)	Changed the word "needed" to "required." System/equipment operability is defined by required components. Added Note 5 as EAL only applies to equipment required for the current operating mode and would be adequately addressed by Tech. Spec. operability requirements.
3	Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).	HU3.3	Movement of personnel within the PLANT PROTECTED AREA is IMPEDED due to an event involving hazardous materials (e.g., a chemical spill or toxic gas release from an area outside the plant PROTECTED AREA).	Added the word "PLANT" to protected area to distinguish from the ISFSI protected area that is located outside the plant protected area. Deleted the term "offsite" to preclude confusing areas outside the Protected Area but on site. Revised the example accordingly.
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 conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles. (Note 7)	DCPP has a very large site (approximately 8 miles from gate to plant Protected Area). Deleted the term "on-site" as site access could be precluded without adverse on-site conditions. 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 DCPP.

Note	EAL #3 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.	N/A	Note 7: This EAL does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.	This note, designated Note #7, is intended to apply to generic example EAL #4, not #3 as specified in the generic guidance.
N/A	N/A	N/A	Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.	Added Note 5 as EAL only applies to equipment required for the current operating mode and would be adequately addressed by Technical Specification operability requirements.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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 minutes of **any** of the following FIRE detection indications (Note 1): • Report from the field (i.e., visual observation). • Receipt of multiple (more than 1) fire alarms or indications. • Field verification of a single fire alarm. **AND** The FIRE is located within **any** Table H-1 area.	Table H-1 provides a tabularized list of 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 associated with	Table H-1 provides a tabularized list of site-specific fire areas. Revised EAL wording, second conditional to read: *"The fire alarm is associated with **any** Table H-1 area"* The fire alarm is not, in and of itself, an indication of fire. The existence of the fire must be verified. It is those fire alarms

	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.		**any** Table H-1 area. **AND** The existence of a FIRE is **not** verified within 30 minutes of alarm receipt. (Note 1)	associated with the Table H-1 fire areas that are of concern. A project to add National Fire Protection Association (NFPA) Standard 805 Incipient Fire Detection is expected to be complete prior to PG&E implementation of NEI 99-01 Rev. 6.
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 ISFSI PROTECTED AREA or PLANT PROTECTED AREA **not** extinguished within 60 minutes of the initial report, alarm or indication. (Note 1)	DCPP has an ISFSI PROTECTED AREA 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 ISFSI or PLANT PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.	DCPP has an ISFSI PROTECTED AREA 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 SM/SEC/ED 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording.

Table H-1 Fire Areas
• Containment • Auxiliary Building • Fuel Handling Building • Turbine Building • Intake Structure Lower Levels • Pipe Rack • Main, Auxiliary & Startup Transformers

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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 SM/SEC/ED warrant declaration of a UE. MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 SM/SEC/ED 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	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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 Watch Commander. **OR** A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.	Example EALs #1 and #2 have been combined into a single EAL for ease of use. The Security Watch Commander 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	DCPP IC#(s)	DCPP IC Wording	Difference Justification
HA5	Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown. MODE: All	HA5	Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown. MODE: 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	Limited mode applicability to the modes specified in Table H-1. Added the following note to the HA5 bases: "***<u>NOTE</u>**: IC HA5 mode applicability has been limited to the applicable modes identified in Table H-2 Safe Operation & Shutdown Rooms/Areas. If due to plant operating procedure or plant configuration changes, the applicable plant modes specified in Table H-2 are changed, a corresponding change to Attachment 3 'Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases' and to IC HA5 mode applicability is required."*

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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(s)**
Auxiliary Building – 115' - BASTs	2, 3, 4
Auxiliary Building – 100' – BA Pumps	2, 3, 4
Auxiliary Building – 85' – Aux Control Board	2, 3, 4
Auxiliary Building – 64' – BART Tank area	2, 3, 4
Area H (below Control Room) – 100' 480V Bus area/rooms	3, 4

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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 requiring plant control to be transferred from the Control Room to the Hot Shutdown Panel area.	Reworded EAL to express cause and effect for control room evacuation. The Hot Shutdown Panel area is the site-specific location of the remote shutdown panels/local control stations.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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 SM/SEC/ED warrant declaration of an ALERT. MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 SM/SEC/ED, 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	DCPP IC#(s)	DCPP IC Wording	Difference 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 distinguish from the ISFSI protected area that is located outside the plant protected area.

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 Watch Commander.	The Security Watch Commander is the site-specific security shift supervision. Added the word "PLANT" to protected area to distinguish from the ISFSI protected area that is located outside the plant protected area.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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 Rev. 6 guidance.**

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 Hot Shutdown Panel area. **AND** Control of **any** of the following key safety functions is **not** reestablished within 15 minutes (Note 1): • Reactivity (Modes 1, 2, and 3 **only**) • Core cooling • RCS heat removal	The Hot Shutdown Panel area are 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 Rev. 6 guidance.**

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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 SM/SEC/ED warrant declaration of a Site Area Emergency. MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 SM/SEC/ED 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	DCPP IC#(s)	DCPP IC Wording	Difference 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 DCPP 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 Rev. 6 guidance.**

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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*] • RCS heat removal	N/A	N/A	IC HG1 and associated example EAL are not implemented in the DCPP 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 Plant 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 EPA PAGs. a. If, for whatever reason, the CR 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

	OR 2. Damage to spent fuel has occurred or is IMMINENT.			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 Rev. 6 and thus HG1 is adequately bounded as described above. **This is an acceptable deviation from the generic NEI 99-01 Rev. 6 guidance.**

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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 SM/SEC/ED warrant declaration of a General Emergency. MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 SM/SEC/ED 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	DCPP IC#(s)	DCPP IC Wording	Difference 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 vital buses for 15 minutes or longer. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	The DCPP vital buses are the site-specific emergency buses.

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 Unit 1 or Unit 2 vital 4.16KV buses F, G and H for ≥ 15 minutes. (Note 1)	4.16KV vital buses 1(2)F, 1(2)G, and 1(2)H 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 SM/SEC/ED 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording.

Table S-1 AC Power Capability		
	Unit 1	Unit 2
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• Aux XFMR 1-2 fed from the Main Generator • DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• Aux XFMR 2-2 fed from the Main Generator • DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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 minutes. (Note 1)	The site-specific Safety System Parameters are listed in Table S-2.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SM/SEC/ED 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 DCPP 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 • In-core TC temperature • Level in at least one SG • Auxiliary or emergency feed flow in at least one SG

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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 permissible 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. Changed the word "allowable" to "permissible" to be consistent with the DCPP Tech. Spec. 3.4.16 terminology.

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference Justification
1	(Site-specific radiation monitor) reading greater than (site-specific value).	SU4.2	With letdown in service, procedurally directed letdown dose point radiation > 3 R/hr.	Initial indication of Fuel Clad degradation can be determined by measuring the external radiation dose rate at a distance of one foot from the center of the letdown line in the letdown heat exchanger room using the technique described in Attachment 7.1 of EP RB-14A, Initial Detection of Core Damage. An external radiation dose rate exceeding 3 R/hr indicates Fuel Clad degradation greater than Tech. Spec. allowable limits.
2	Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications.	SU4.1	RCS activity > Technical Specification Section 3.4.16 permissible limits.	Deleted the words "Sample analysis indicates…" because reactor coolant activity is only determined by sample analysis. Changed 'reactor coolant activity" to "RCS activity" to conform to site specific terminology. DCPP Tech. Spec. Section 3.4.16 provides the Tech. Spec. permissible coolant activity limits. Changed the word "allowable" to "permissible" to be consistent with the DCPP Tech. Spec. 3.4.16 terminology.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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 minutes. **OR** RCS identified leakage > 25 gpm for ≥ 15 minutes. **OR** Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 minutes. (Note 1)	Example EALs #1, 2, and 3 have been combined into a single EAL for usability.
2	RCS identified leakage greater than (site-specific value) for 15 minutes or longer.			
3	Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer.			
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SM/SEC/ED 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 DCPP EAL scheme by referencing the “time limit” specified within the EAL wording.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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** RTS setpoint is exceeded. **AND** A subsequent automatic trip or manual trip action taken at the control room panels (CC1, VB2 or VB5) 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** Reactor Trip System (RTS) setpoint is exceeded" to clarify that it is a failure of the automatic trip when a valid trip signal has been exceed. CR panels CC1, VB2, or VB5 are the reactor control consoles where an immediate manual reactor trip can be initiated.
2	a. A manual trip ([PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. **EITHER** of the following: 1. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor. **OR**	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 control room panels (CC1, VB2 or VB5) 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** 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. CR panels CC1, VB2, or VB5 are the reactor control consoles where an immediate manual reactor trip can be initiated. Combined conditions b.1 and b.2 into a single statement to simplify

	2 A subsequent automatic (trip [PWR] / scram [BWR]) is successful in shutting down the reactor.			the presentation.
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	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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 offsite 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 for simplification of presentation. Table S-4 provides a site-specific list of onsite, offsite (ORO), and NRC communications methods.
2	Loss of **ALL** of the following ORO communications methods: (site-specific list of communications methods)			
3	Loss of **ALL** of the following NRC communications methods: (site-specific list of communications methods)			

Table S-4 Communication Methods			
System	**Onsite**	**Offsite**	**NRC**
Unit 1, Unit 2 and TSC Radio Consoles	X	X	
DCPP Telephone System (PBX)	X	X	X
Portable radio equipment (handie-talkies)	X		
Operations Radio System	X	X	
Security Radio Systems	X		
CAS and SAS Consoles	X	X	X
Fire Radio System	X		
Hot Shutdown Panel Radio Consoles	X	X	X
Public Address System	X		
NRC FTS			X
Mobile radios	X		
Satellite phones	X	X	X
Direct line (ATL) to the County and State OES		X	

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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	**EITHER**: **Any** penetration is not isolated within 15 minutes of a VALID containment isolation signal. (Note 1) **OR** Containment pressure ≥ 22 psig with < one full train of containment depressurization equipment operating per design for ≥ 15 minutes. (Notes 1, 9)	Example EALs #1 and #2 have been combined into a single EAL. 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 (22 psig) is the pressure at which the containment depressurization equipment should actuate and begin performing its function. Added Note 9 to specify what constitutes a full train of containment depressurization equipment.
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 SM/SEC/ED should declare the event promptly upon	Added Note 1 to be consistent in its use for EAL thresholds with a timing component.

			determining that time limit has been exceeded, or will likely be exceeded.	
N/A	N/A	N/A	Note 9: One Containment Spray pump and two CFCUs comprise one full train of depressurization equipment.	Added Note 9 to specify what constitutes a full train of containment depressurization equipment.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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 vital buses for 15 minutes or longer. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	The DCPP vital buses are the site-specific emergency buses.

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 Unit 1 or Unit 2 vital 4.16KV buses F, G, and H reduced to a single power source for ≥ 15 minutes. (Note 1) **AND** A failure of that single power source will result in loss of **all** AC power to SAFETY SYSTEMS.	4.16KV vital buses 1(2)F, 1(2)G, and 1(2)H are the site-specific emergency buses. Site-specific AC power sources are listed in Table S-1. Reworded the second condition for clarity.
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 SM/SEC/ED 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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	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 minutes. (Note 1) **AND** **Any** significant transient is in progress, Table S-3.	The site-specific Safety System Parameters are listed in Table S-2. The site-specific significant transients are listed in Table S-3. DCPP is a PWR and thus does not include thermal power oscillations > 10%.

	greater than 10% [*BWR*]			
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The SM/SEC/ED 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 DCPP 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 • In-core TC temperature • Level in at least one SG • Auxiliary or emergency feed flow in at least one SG

Table S-3 Significant Transients
• Reactor trip • Runback > 25% thermal power • Electrical load rejection > 25% electrical load • ECCS actuation

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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 consoles are not successful in shutting down the reactor. MODE: 1 - Power Operation	None

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 control room panels (CC1, VB2, or VB5) 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. CR panels CC1, VB2, or VB5 are the reactor control consoles where an immediate manual reactor trip can be initiated.
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	DCPP IC#(s)	DCPP IC Wording	Difference Justification
SA9	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SA9.1	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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. Tsunami has been added as a site-specific hazard for DCPP.

Table S-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or TORNADO strike • FIRE • EXPLOSION • Tsunami • Other events with similar hazard characteristics as determined by the SM/SEC/ED

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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 vital buses for 15 minutes or longer. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	The DCPP vital buses are the site-specific emergency buses.

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 capability, Table S-1, to Unit 1 or Unit 2 vital 4.16KV buses F, G, and H for ≥ 15 minutes. (Note 1)	4.16KV vital buses 1(2)F, 1(2)G, and 1(2)H 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 SM/SEC/ED 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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 conditions met. • CSFST Heat Sink RED path conditions met. **AND** Bleed and feed criteria 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. For DCPP indication that heat removal is extremely challenged is manifested by CSFST Heat Sink RED path conditions met in combination with bleed and feed criteria being met. Refer to Attachment A for justification of incorporation for the bleed and feed condition threshold associated with loss of heat sink.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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	DCPP EAL #	DCPP EAL Wording	Difference 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** 125 VDC power based on battery bus voltage indications < 105 VDC on **all** Unit 1 or Unit 2 vital DC buses for ≥ 15 minutes. (Note 1)	105 VDC is the site-specific minimum vital DC bus voltage.
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 SM/SEC/ED 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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 vital buses. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	The DCPP vital buses are the site-specific emergency buses.

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 capability, Table S-1, to Unit 1 or Unit 2 vital 4.16KV buses F, G, and H. **AND EITHER:** • Restoration of at least one 4.16KV vital bus in < 4 hours is **not** likely. (Note 1) • CSFST Core Cooling RED path conditions met.	4.16KV vital buses 1(2)F, 1(2)G, and 1(2)H are the site-specific emergency buses. Site-specific AC power sources are tabularized in Table S-1. 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 SM/SEC/ED 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 DCPP EAL scheme by referencing the “time limit” specified within the EAL wording.

NEI IC#	NEI IC Wording	DCPP IC#(s)	DCPP IC Wording	Difference 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 IC SG8 has been grouped under the loss of vital DC category 2. The DCPP vital buses are the site-specific emergency buses.

NEI Ex. EAL #	NEI Example EAL Wording	DCPP EAL #	DCPP EAL Wording	Difference 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 capability, Table S-1, to Unit 1 or Unit 2 vital 4.16KV buses F, G, and H for ≥ 15 minutes. **AND** Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **all** Unit 1(2) vital DC buses for ≥ 15 minutes. (Note 1)	4.16KV vital buses 1(2)F, 1(2)G, and 1(2)H are the site-specific emergency buses. Site-specific AC power sources are tabularized in Table S-1. 105 VDC is the site-specific minimum vital DC bus voltage.
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 SM/SEC/ED 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 DCPP EAL scheme by referencing the "time limit" specified within the EAL wording.

Attachment A

Bleed and Feed Condition Associated with CSFST Heat Sink Red Path Thresholds

This discussion is to be used for the following EALs entry conditions:

- Table F-1 Fuel Clad Barrier Potential Loss B.2 "CSFST Heat Sink RED Path conditions met and Heat Sink required."
- Table F-1 Reactor Coolant System (RCS) Barrier Potential Loss B.12 "CSFST Heat Sink RED Path conditions met and Heat Sink required."
- SS6.1 "CSFST heat Sink RED Path conditions met."

**

The NEI 99-01 Rev. 6 document, in describing the entry condition of "CSFST Heat Sink RED Path conditions met," states that "This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., <u>loss of an effective secondary-side heat sink</u>)."

Merely meeting the CSFST Heat Sink Red Path for Loss of Heat Sink does not meet this condition. In order to do so, an additional trigger has been added that states "Bleed and feed conditions have been met."

The Bleed and Feed entry condition is based on Westinghouse analysis, and detailed out in the Loss of Heat Sink procedure EOP FR H-1. At DCPP, the Bleed and Feed entry conditions are the indication of a <u>loss of the effective secondary side heat sink</u> that is stated in NEI 99-01 Rev. 6. The indication of a loss of the effective secondary side heat sink is continuously monitored as a fold out page item while in the Loss of Heat Sink procedure. Currently Bleed and Feed is initiated if the following condition is met:

- WR SG Level in any 3 SGs < 18%, AND all NR SG Levels are < 15%

This is the condition that meets NEI 99-1 Rev. 6 description of an "extreme challenge to the ability to remove RCS heat using the steam generators" wherein there is a "<u>loss of an effective secondary-side heat sink.</u>"

**

The procedure background document titled "Westinghouse Owners Group (WOG) Emergency Response Guideline (ERG) FR-H.1 "Response to Loss of Secondary Heat Sink, HP-Rev. 3, March 31, 2014"" was reviewed.

This document states, in part, that "The objective of guideline FR-H.1 is to maintain reactor coolant system (RCS) heat removal capability by establishing feed flow to a steam generator or by establishing RCS bleed and feed heat removal. Guideline FR-H.1 is entered at the <u>first indication</u> that secondary heat removal capability <u>may</u> be challenged. <u>This permits maximum time for operator action to restore feed water flow</u> to at least one steam generator <u>before secondary inventory is depleted and secondary heat removal capability is lost</u>. Once secondary heat removal capability is lost, RCS bleed and feed must be established to minimize core uncovery and <u>prevent</u> an inadequate core cooling condition."

Note that the analysis is specifically devoid of verbiage that any fission product barriers are potentially or actually being challenged.

The WOG ERG goes on to state that "The initial RCS depressurization after reactor trip gives way to a <u>quasi-steady state period characterized by core decay heat energy removal through the steam generators</u>. As secondary side mass is depleted through the condenser steam dumps, steam generator PORVs or steam generator safety valves, the steam generators will slowly dry out. <u>During this period the RCS pressure and temperature will be relatively constant</u> as the steam generator level continues to decrease and more of the steam generator tube heat transfer area uncovers. <u>There will still be sufficient secondary heat removal capability, even with a portion of the tubes uncovered, to maintain relatively stable RCS conditions for pressure, temperature and pressurizer level.</u>"

**

This methodology is acceptable based on NEI 99-01 Rev. 6 developer notes that state:

From page 79 of NEI 99-01 Rev. 6, in the Developer Notes:

3. The fission product barrier thresholds specified within a scheme are expected to reflect *plant-specific design and operating characteristics*. This may require that developers create different thresholds than those provided in the generic guidance.

From page 102 of NEI 99-01 Rev. 6, in the Developer Notes:

The CSFST thresholds may be addressed in one of 3 ways:

1) Not incorporated; thresholds will use parameters and values as discussed in the Developer Notes.

2) *Incorporated along with parameter and value thresholds* (e.g., a fuel clad loss would have 2 thresholds such as "CETs > 1200°F" and "Core Cooling Red entry conditions met."

3) Used in lieu of parameters and values for all thresholds.

Enclosure
Attachment 2
PG&E Letter DCL-16-099

EAL Technical Basis Document Markup

Appendix D - Emergency Action Level Technical Basis Document

9/27/16

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 Diablo Canyon Power Plant (DCPP). It should be used to facilitate review of the DCPP EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EP G-1 Emergency Classification and Emergency Plan Activation, may use this document as a technical reference in support of EAL interpretation. This information may assist the SM/SEC/ED 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.

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

2.0 DISCUSSION

2.1 Background

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

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

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

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

Subsequently, Revision 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 (ref. 4.1.1), DCPP 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 (CMT): 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

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 DCPP EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
  - EALs applicable under <u>any</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, 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 DCPP EAL categories are aligned to and represent the NEI 99-01"Recognition Categories." Subcategories are used in the DCPP scheme as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The DCPP EAL categories and subcategories are listed below.

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
R – Abnormal **R**ad Levels / Rad Effluent	1 – Radiological Effluent 2 – 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 – SM/SEC/ED 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 – RTS 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 Wall Chart. The user of the EAL Classification Wall Chart 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 (Any, Hot, Cold), EAL category (R, C, H, S, E and F) and EAL subcategory. A summary explanation of each category and subcategory is given at the beginning of the technical bases discussions of the EALs included in the category. For each EAL, the following information is provided:

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:

An EAL basis section that provides both generic and site-specific ERO decision making guidance as well as background information that supports the rationale for the EAL as provided in NEI 99-01 Rev. 6.

DCPP 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 °F with all reactor vessel head closure bolts fully tensioned

5 Cold Shutdown

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

6 Refueling

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

D Defueled

Reactor vessel contains no irradiated fuel (full core off-load during refueling or extended outage).

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

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Shift Manager/Site Emergency Coordinator/Emergency Director (SM/SEC/ED) 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 (ref. 4.1.9).

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 SM/SEC/ED should not wait until the applicable time has elapsed. The SM/SEC/ED 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 cannot be determined, 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). For example, a coolant activity sample is taken. Chemistry reports results indicate activity greater than Technical Specification limits. The classification clock begins when Chemistry reports the sample results.

3.1.6 SM/SEC/ED 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 SM/SEC/ED 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 SM/SEC/ED 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.” (ref. 4.1.9).

3.2.1 Classification of Multiple Events and Conditions

When multiple emergency events or conditions are present, the user will recognize 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.

For example, a loss of decay heat removal when in Mode 5 results in RCS temperature exceeding 200°F. Escalation of the loss of decay heat removal event will be via the cold condition mode EALs even though the plant is now in Mode 4 as a result of the RCS temperature increase. However, any subsequent new event/condition must be assessed against the hot condition EALs (Mode 4 and above).

3.2.3 Classification of Imminent Conditions

Although EALs provide specific thresholds, the SM/SEC/ED 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 SM/SEC/ED, 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. Refer to EP G-1 Emergency Classification and Emergency Plan Activation for guidance on downgrading and terminating an ECL. Refer to EP OR-3 Emergency Recovery for guidance for entering long-term recovery.

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 preclude 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 SM/SEC/ED completing the review and steps necessary to make the emergency declaration. This provision is included to ensure that any public protective actions resulting from the emergency classification are truly warranted by the plant conditions.

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

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

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 (ref. 4.1.3) is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR § 50.72 (ref. 4.1.4) within one hour of the discovery of the undeclared event or condition (refer to XI1.ID2 Regulatory Reporting Requirements and

Reporting Process (ref. 4.1.11)). 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 10CFR 50.72 Immediate Notification Requirements for Operating Nuclear Power Reactors

4.1.5 10CFR 50.73 License Event Report System

4.1.6 Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points

4.1.7 Technical Specifications Table 1.1-1 Modes

4.1.8 Administrative Procedure AD8.DC54 "Containment Closure"

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

4.1.10 DCPP Emergency Plan

4.1.11 XI1.ID2 Regulatory Reporting Requirements and Reporting Process

4.1.12 DCPP Security and Safeguards Contingency Plan

4.2 Implementing

4.2.1 EP G-1 Emergency Classification and Emergency Plan Activation

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

4.2.3 DCPP EAL Wall Chart

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions

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 process, 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.

Confinement Boundary

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

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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 "Containment Closure" (ref. 4.1.8).

Degraded Performance

As applied to hazardous event thresholds, damage significant enough to cause concern regarding the operability or reliability of the affected safety system train.

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 Protective Action Guidelines (EPA PAG)

The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs requires DCPP 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. (*refer to Section 2.2*)

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 process 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 DCPP 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 DCPP. 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.

ISFSI Protected Area

Areas within the ISFSI to which access is strictly controlled in accordance with the station's Security Plan.

~~**Normal Levels**~~

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

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) (ref. 4.1.8).

Owner Controlled Area (OCA)

For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan (ref. 4.1.12).

Projectile

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

Plant Protected Area

Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the Plant Protected Area.

RCS Leakage

RCS Leakage shall be:

a. Identified Leakage
   1. Leakage, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leak off), that is captured and conducted to collection systems or a sump or collecting tank;
   2. 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;
   3. Reactor Coolant System (RCS) leakage through a steam generator to the secondary system (primary to secondary leakage).

b. Unidentified Leakage

   All leakage (except RCP seal water injection or leak off) that is not identified leakage.

c. Pressure Boundary Leakage

   Leakage (except primary to secondary leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

d. 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).

Reduced Inventory Condition (RIC)

The condition existing whenever RCS water level is lower than 3 feet below the reactor vessel flange (below 111-foot elevation) with fuel in the core.

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.

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.

Security Event

Any incident representing an attempted, threatened, or actual breach of the security system or reduction of the operational effectiveness of that system. A security event can result in either a SECURITY CONDITION or HOSTILE ACTION (ref. 4.1.12).

Site Area Emergency

Events are in process 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 Boundary

As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points (ref. 4.1.6).

Tornado

A violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

Unisolable

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

NOTE: If an isolation valve could not be accessed due to local conditions (i.e. high radiation, temperature, etc.) then that would also make a leak unisolable, even though the inaccessible valve could isolate the leak.

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 process 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.

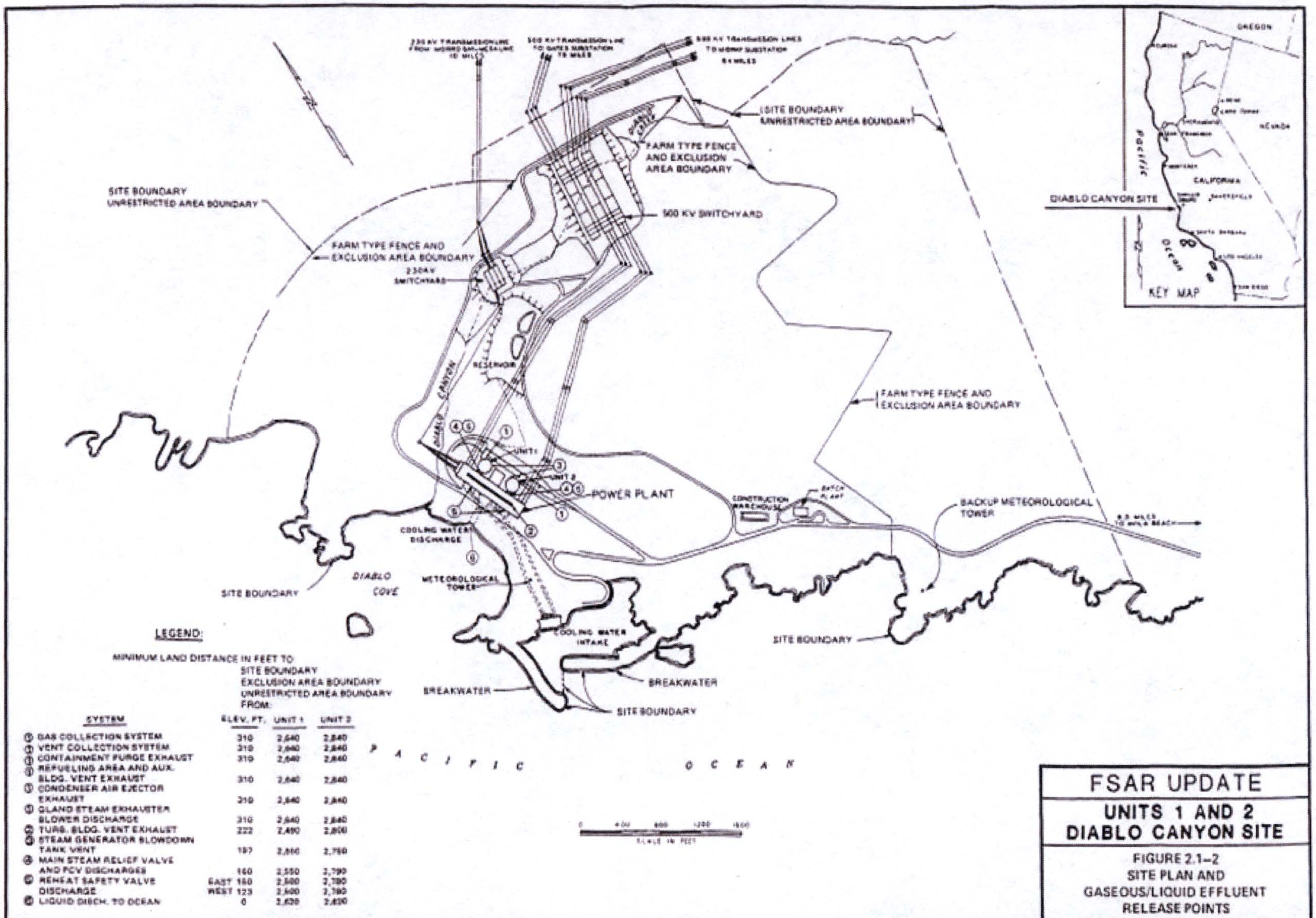
No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

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. 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.



FSAR UPDATE

UNITS 1 AND 2 DIABLO CANYON SITE

FIGURE 2.1–2 SITE PLAN AND GASEOUS/LIQUID EFFLUENT RELEASE POINTS

Revision 11 November 1996

5.2 Abbreviations/Acronyms

°F ..... Degrees Fahrenheit
° ..... Degrees
AC ..... Alternating Current
AFW ..... Auxiliary Feedwater
AOP ..... Abnormal Operating Procedure
ATL ..... Automatic Tie Line
ATWS ..... Anticipated Transient Without Scram
BA ..... Boric Acid
BART ..... Boric Acid Reserve Tank
BAST ..... Boric Acid Storage Tank
CAS ..... Central Alarm Station
CCW ..... Component Cooling Water
CC(#) ..... Control Console (number)
CDE ..... Committed Dose Equivalent
CEDE ..... Committed Effective Dose Equivalent
CET ..... Core Exit Thermocouple
CFCU ..... Containment Fan Cooling Unit
CFR ..... Code of Federal Regulations
CMT ..... Containment
CSFST ..... Critical Safety Function Status Tree
CST ..... Condensate Storage Tank
DBA ..... Design Basis Accident
DC ..... Direct Current
DCPP ..... Diablo Canyon Power Plant
DDE ..... Double Design Earthquake
DE ..... Design Earthquake
EAL ..... Emergency Action Level
ECCS ..... Emergency Core Cooling System
ECL ..... Emergency Classification Level
ED ..... Emergency Director
EDE ..... Effective Dose Equivalent
EFM ..... Earthquake Force Monitor
ENF ..... Emergency Notification Form
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

ERFDS .... Emergency Response Facility Display System
FAA .... Federal Aviation Administration
FBI .... Federal Bureau of Investigation
FEMA .... Federal Emergency Management Agency
FSAR .... Final Safety Analysis Report
ft. .... Feet
FTS .... Federal Telephone System
GDC .... General Design Criteria
GE .... General Emergency
HASP .... High Alarm Setpoint
HOO .... NRC Headquarters Operations Officer
IC .... Initiating Condition
in. .... Inches
IPEEE .... Individual Plant Examination of External Events (Generic Letter 88-20)
K_{eff} .... Effective Neutron Multiplication Factor
LCO .... Limiting Condition of Operation
LER .... Licensee Event Report
LOCA .... Loss of Coolant Accident
LWR .... Light Water Reactor
MEDT .... Miscellaneous Equipment Drain Tank
MPC .... Maximum Permissible Concentration
MPC .... Multi-Purpose Canister
mR, mRem, mrem, mREM .... milli-Roentgen Equivalent Man
MSL .... Main Steam Line
MW .... Megawatt
NEI .... Nuclear Energy Institute
NEIC .... National Earthquake Information Center
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
NUREG .... Nuclear Regulation
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM .... Off-site Dose Calculation Manual
OES .... Office of Emergency Services
ORO .... Offsite Response Organization
PA .... Protected Area

PAG	Protective Action Guideline
PAM	Post Accident Monitoring
PAR	Protective Action Recommendation
PBX	Private Branch Exchange
PGA	Peak Ground Acceleration
PPC	Plant Process Computer
PRA/PSA	Probabilistic Risk Assessment / Probabilistic Safety Assessment
PRT	Pressurizer Relief Tank
PSIG	Pounds per Square Inch Gauge
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
R	Roentgen
RCC	Reactor Control Console
RCDT	Reactor Coolant Drain Tank
RCS	Reactor Coolant System
RHR	Residual Heat Removal
Rem, rem, REM	Roentgen Equivalent Man
RETS	Radiological Effluent Technical Specifications
RTS	Reactor Trip System
R(P)V	Reactor (Pressure) Vessel
RVLIS	Reactor Vessel Level Indicating System
RVRLIS	Reactor Vessel Refueling Level Indicating System
RWST	Refueling Water Storage Tank
SAE	Site Area Emergency
SAMG	Sever Accident Management Guideline
SAR	Safety Analysis Report
SAS	Secondary Alarm Station
SBO	Station Blackout
SCBA	Self-Contained Breathing Apparatus
SCMM	Sub Cooled Margin Monitor
SEC	Site Emergency Coordinator
SG	Steam Generator
SI	Safety Injection
SM	Shift Manager
SPDS	Safety Parameter Display System
SRO	Senior Reactor Operator
SSF	Safe Shutdown Facility
TC	Thermocouple
TEDE	Total Effective Dose Equivalent
TOAF	Top of Active Fuel

TSC .... Technical Support Center
UE .... Unusual Event
UFSAR .... Updated Final Safety Analysis Report
USGS .... United States Geological Survey
VB(#) .... Vertical Board (number)
VDC .... Volts Direct Current
WOG .... Westinghouse Owners Group
WR .... Wide Range
XFMR .... Transformer

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

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

DCPP	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

DCPP	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	1
CS1.2	CS1	2
CS1.3	CS1	3
CG1.1	CG1	1
CG1.2	CG	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

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

DCPP	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
SA9.1	SA9	1
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1
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

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.

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.1 Unusual Event

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

Note 1: The SM/SEC/ED 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 **cannot** be determined, 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	Plant Vent	1(2)-RM-14/14R	-----	2.5E+6 cpm	2.5E+5 cpm	8.0E+4 cpm
				5.6E-2 μCi/cc	5.6E-3 μCi/cc	1.8E-3 μCi/cc
		1(2)-RM-87	1.9E-10 amps	-----	-----	-----
			3.2E-1 μCi/cc			
Liquid	Liquid Radwaste Effluent Line	0-RM-18	-----	-----	-----	1.6E+5 cpm
	SGBD Tank	1(2)-RM-23	-----	-----	-----	2.0E+4 cpm

Mode Applicability:

All

Definition(s):

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

Basis:

ERO Decision Making Information

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.

Even if a release does not meet the levels of this EAL, a release may be reportable. In these cases, consult Admin Procedure XI1.ID2.

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.

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

Background

The column "UE" gaseous and liquid release values in Table R-1 represent two times the appropriate Offsite Dose Calculation Manual 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.

~~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.).~~

DCPP Basis Reference(s):

1. DCPP Radiological Effluent Technical Specifications
2. EP-CALC-DCPP-1601 Radiological Effluent EAL Values
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 Offsite Dose Calculation Manual 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 Offsite Dose Calculation Manual limits for ≥ 60 minutes. (Notes 1, 2)	

Note 1: The SM/SEC/ED 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 **cannot** be determined, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

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

NOTE: If an isolation valve could not be accessed due to local conditions (i.e. high radiation, temperature, etc.) then that would also make a leak unisolable, even though the inaccessible valve could isolate the leak.

Basis:

ERO Decision Making Information

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

Sample analysis results relative to Offsite Dose Calculation Manual limits are provided by Chemistry.

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

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

Background

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.~~

~~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).~~

DCPP Basis Reference(s):

1. DCPP Radiological Effluent Technical Specifications
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 minutes. (Notes 1, 2, 3, 4)

Note 1: The SM/SEC/ED 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 **cannot** be determined, 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 **only** 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	Plant Vent	1(2)-RM-14/14R	-----	2.5E+6 cpm	2.5E+5 cpm	8.0E+4 cpm
				5.6E-2 µCi/cc	5.6E-3 µCi/cc	1.8E-3 µCi/cc
		1(2)-RM-87	1.9E-10 amps	----	----	----
			3.2E-1 µCi/cc			
Liquid	Liquid Radwaste Effluent Line	0-RM-18	-----	-----	-----	1.6E+5 cpm
	SGBD Tank	1(2)-RM-23	-----	-----	-----	2.0E+4 cpm

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

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:

ERO Decision Making Information

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 thyroid CDE

The results from a dose assessment are preferred using actual meteorology, when available. Until available, the pre-calculated effluent monitor values presented in Table R-1 should be used for emergency classification. Once an accurate dose assessment is performed, classification should be based on dose assessment only and not using the effluent monitor values in Table R-1. For example, a Table R-1 Alert effluent threshold is exceeded. However, real-time dose assessment results are available indicating offsite doses less than EAL RA1.2 thresholds. Declaration of an Alert due to EAL RA1.1 is not required.

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.

Background

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.

DCPP Basis Reference(s):

1. EP-CALC-DCPP-1601 Radiological Effluent EAL 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 **only** be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

ERO Decision Making Information

Dose projections are performed by computer-based method (ref. 1, 2, 3). Dose assessments may utilize real-time dose projections and/or field monitoring results.

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.

Background

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.

DCPP Basis Reference(s):

1. EP RB-9, Calculation of Release Rate
2. EP RB-11, Emergency Offsite Dose Calculations
3. EP R-2, Release of Airborne Radioactive Materials Initial Assessment
4. 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 minutes of exposure. (Notes 1, 2)	

Note 1: The SM/SEC/ED 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 **cannot** be determined, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

ERO Decision Making Information

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

Background

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.~~

DCPP Basis Reference(s):

1. EP R-3 Release of Radioactive Liquids
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 are > 10 mR/hr and are expected to continue for ≥ 60 minutes. • Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 minutes of inhalation. (Notes 1, 2)

Note 1: The SM/SEC/ED 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 **cannot** be determined, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

ERO Decision Making Information

EP RB-8, Instructions for Field Monitoring Teams provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

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

Background

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.~~

DCPP Basis Reference(s):

1. EP RB-8, Instructions for Field Monitoring Teams
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 minutes. (Notes 1, 2, 3, 4)

Note 1: The SM/SEC/ED 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 **cannot** be determined, 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 **only** 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	Plant Vent	1(2)-RM-14/14R	-----	2.5E+6 cpm	2.5E+5 cpm	8.0E+4 cpm
				5.6E-2 μCi/cc	5.6E-3 μCi/cc	1.8E-3 μCi/cc
		1(2)-RM-87	1.9E-10 amps	----	----	----
			3.2E-1 μCi/cc			
Liquid	Liquid Radwaste Effluent Line	0-RM-18	-----	-----	-----	1.6E+5 cpm
	SGBD Tank	1(2)-RM-23	-----	-----	-----	2.0E+4 cpm

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

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:

ERO Decision Making Information

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 thyroid CDE

The results from a dose assessment are preferred using actual meteorology, when available. Until available, the pre-calculated effluent monitor values presented in Table R-1 should be used for emergency classification. Once an accurate dose assessment is performed, classification should be based on dose assessment only and not using the effluent monitor values in Table R-1. For example, a Table R-1 SAE effluent threshold is exceeded. However, real-time dose assessment results are available indicating offsite doses less than EAL RS1.2 thresholds. Declaration of a Site Area Emergency due to EAL RS1.1 is not required.

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.

Background

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 thyroid CDE) (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.

DCPP Basis Reference(s):

1. EP-CALC-DCPP-1601 Radiological Effluent EAL 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 **only** be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

ERO Decision Making Information

Dose projections are performed by computer-based method (ref. 1, 2, 3). Dose assessments may utilize real-time dose projections and/or field monitoring results.

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

Background

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.~~

DCPP Basis Reference(s):

1. EP RB-9, Calculation of Release Rate
2. EP RB-11, Emergency Offsite Dose Calculations
3. EP R-2, Release of Airborne Radioactive Materials Initial Assessment
4. 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 are > 100 mR/hr and are expected to continue for ≥ 60 minutes.
- Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 minutes of inhalation.

(Notes 1, 2)

Note 1: The SM/SEC/ED 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 **cannot** be determined, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

ERO Decision Making Information

EP RB-8, Instructions for Field Monitoring Teams provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

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

Background

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.~~

DCPP Basis Reference(s):

1. EP RB-8, Instructions for Field Monitoring Teams
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 minutes. (Notes 1, 2, 3, 4)

Note 1: The SM/SEC/ED 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 **cannot** be determined, 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 **only** 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	Plant Vent	1(2)-RM-14/14R	-----	2.5E+6 cpm	2.5E+5 cpm	8.0E+4 cpm
Gaseous	Plant Vent	1(2)-RM-14/14R	-----	5.6E-2 μCi/cc	5.6E-3 μCi/cc	1.8E-3 μCi/cc
Gaseous	Plant Vent	1(2)-RM-87	1.9E-10 amps	----	----	----
Gaseous	Plant Vent	1(2)-RM-87	3.2E-1 μCi/cc	----	----	----
Liquid	Liquid Radwaste Effluent Line	0-RM-18	-----	-----	-----	1.6E+5 cpm
Liquid	SGBD Tank	1(2)-RM-23	-----	-----	-----	2.0E+4 cpm

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

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:

ERO Decision Making Information

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 thyroid CDE

The results from a dose assessment are preferred using actual meteorology, when available. Until available, the pre-calculated effluent monitor values presented in Table R-1 should be used for emergency classification. Once an accurate dose assessment is performed, classification should be based on dose assessment only and not using the effluent monitor values in Table R-1. For example, a Table R-1 GE effluent threshold is exceeded. However, real-time dose assessment results are available indicating offsite doses less than EAL RG1.2 thresholds. Declaration of a General Emergency due to EAL RG1.1 is not required.

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.

Background

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 thyroid CDE) (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.

DCPP Basis Reference(s):

1. EP-CALC-DCPP-1601 Radiological Effluent EAL 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 **only** be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

<u>ERO Decision Making Information</u>

<u>Dose projections are performed by computer-based method (ref. 1, 2, 3). Dose assessments may utilized real-time dose projections and/or field monitoring results.</u>

<u>Background</u>

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.~~

DCPP Basis Reference(s):

1. EP RB-9, Calculation of Release Rate

2. EP RB-11, Emergency Offsite Dose Calculations
3. EP R-2, Release of Airborne Radioactive Materials Initial Assessment
4. 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 are > 1,000 mR/hr and are expected to continue for ≥ 60 minutes.
- Analyses of field survey samples indicate thyroid CDE > 5,000 mrem for 60 minutes of inhalation.

(Notes 1, 2)

Note 1: The SM/SEC/ED 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 **cannot** be determined, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

ERO Decision Making Information

EP RB-8, Instructions for Field Monitoring Teams provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

Background

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.~~

DCPP Basis Reference(s):

1. EP RB-8, Instructions for Field Monitoring Teams
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 equivalent indication.

AND

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

- RM-58 Spent Fuel Pool Area
- RM-59 New Fuel Area
- RM-2 Containment Area (Mode 6 **only**)
- **Any** temporary installed monitor in vicinity of the REFUELING PATHWAY (when installed)

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:

ERO Decision Making Information

Note that this EAL is applicable only in cases where the elevated reading is due to an UNPLANNED loss of water level.

The Spent Fuel Pool (SFP) low water level alarm setpoint is 23 ft. 9 in. above the top of irradiated fuel seated in the SFP storage racks or 137 feet 4 inches elevation.

The Refueling Cavity low water level alarm setpoint is at 138 feet elevation as measured on Reactor Vessel Refueling Level Indicating System (RVRLIS) (i.e., 24 feet above the top of reactor vessel flange).

The reading on an area radiation monitor (permanently installed or temporary) located near the Reactor Cavity may increase due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Elevated radiation monitor indications to the low alarm setpoint will need to be combined with another indicator (or personnel report) of water loss (ref. 5, 6)

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.

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.

Background

SFP water level at 136 feet 7 inches elevation is the Technical Specification LCO limit (SR 3.7.15) that requires 23 ft. of water above irradiated fuel seated in the Spent Fuel Pool storage racks.

A minimum depth of 23 feet of water over the irradiated fuel assemblies in the SFP and 23 feet of water over the reactor vessel flange in the refueling cavity is maintained to ensure sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits and to ensure that the offsite dose consequences due to a postulated fuel handling accident are acceptable (ref. 1, 2, 3, 4).

Loss of Spent Fuel Pool water inventory results from either a rupture of the pool or transfer canal liner, or failure of the spent fuel cooling system and the subsequent boil-off. Allowing SFP water level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment. The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 feet above the top of the reactor vessel flange.

While a radiation monitor (RM-58, RM-59, RM-2 or temporarily installed monitors in the vicinity of the REFUELING PATHWAY) could detect an increase in dose due to a drop in the water level, it might not be a reliable indication, in and of itself, of whether or not there is adequate shielding from irradiated fuel (ref. 5, 6).

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.~~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.~~

DCPP Basis Reference(s):

1. Technical Specification 3.7.15, SFP Level
2. Technical Specification 3.9.7, Refueling Cavity Water Level
3. AR PK11-04 input 1064, Spent Fuel Pool Lvl/Temp
4. AR PK02-22 input 1185, Rx Vsl Refueling Lvl (red)
5. OP AP-22, Spent Fuel Pool Abnormalities
6. AR PK-11-10, FHB High Radiation
7. 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	**Unusual Event**
Uncovery of irradiated fuel in the REFUELING PATHWAY.	

Mode Applicability:

All

Definition(s):

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

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

Basis:

ERO Decision Making Information

This ~~IC~~ EAL addresses events that have caused ~~imminent or actual damage to an irradiated fuel assembly, or~~ a significant lowering of water level within the ~~spent fuel pool~~REFUELING PATHWAY ~~*(see Developer Notes)*~~.

This ~~IC~~ EAL applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with ~~IC E-HU1~~EU1.1.

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

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

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

Background

~~EAL #1~~

These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

This EAL escalates from ~~AU2~~ 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.

~~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.~~

DCPP Basis Reference(s):

1. OP AP-21, Irradiated Fuel Damage
2. OP AP-22, Spent Fuel Pool Abnormalities
3. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity.

AND

High alarm on **any** of the following radiation monitors:

- RM-59 New Fuel Storage Area
- RM-58 Spent Fuel Pool Area
- **Any** temporary installed monitor in vicinity of the REFUELING PATHWAY (when installed)
- RM-2 Containment Area (Mode 6 **only**)
- RM-44A/B Containment Ventilation Exhaust (Mode 6 **only**)

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 DCPP ISFSI, the confinement boundary is defined to be the Multi-Purpose Canister (MPC).

Basis:

ERO Decision Making Information

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

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

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

Background

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

The bases for the SFP area radiation high alarms and containment area and ventilation radiation high alarms are a spent fuel handling accident and are, therefore, appropriate for this EAL. In the fuel handling building, a fuel assembly could be dropped in the fuel transfer canal or in the SFP. Should a fuel assembly be dropped in the fuel transfer canal or in the spent fuel pool and release radioactivity above a prescribed level, the area radiation monitors sound an alarm, alerting personnel to the problem. Area radiation monitors in the fuel handling building isolate the normal fuel handling building ventilation system and automatically initiate the recirculation and filtration systems. (ref. 1, 2, 3).

This ~~IC~~ EAL addresses events that have caused IMMINENT or actual damage to an irradiated fuel assembly~~, or a significant lowering of water level within the spent fuel pool *(see Developer Notes)*~~.

~~———~~These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.~~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.~~

~~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)*.~~

DCPP Basis Reference(s):

1. OP AP-21, Irradiated Fuel Damage
2. OP AP-22, Spent Fuel Pool Abnormalities
3. I&C RMS Data Book
4. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.3	**Alert**
Lowering of spent fuel pool level to 10 ft. above top of the fuel racks (Level 2).	

Mode Applicability:

All

Definition(s):

None

Basis:

ERO Decision Making Information

This ~~IC~~ EAL addresses ~~events that have caused imminent or actual damage to an irradiated fuel assembly, or~~ a significant lowering of water level within the spent fuel pool ~~*(see Developer Notes)*~~.

For DCPP Plant SFP Level 2 is 10 ft. (plant El. 123′ 11″) as indicated on LI-801. Backup indication is also available on LI-802. The PPC point for SFP level is L0690A for both units.

Main Annunciator window PK11-04 will alarm at SFP Level 2 (ref. 3).

Escalation of the emergency classification level would be via one or more EALs under ICs ~~AS1~~ RS1 or ~~or AS2~~RS2 ~~*(see AS2 Developer Notes)*~~.

Background

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.

Post-Fukushima order EA-12-051 (ref. 1) required the installation of reliable SFP level indication capable of identifying normal minimum level (Level 1 – 134' 5"), SFP level 10 ft. above the top of the fuel racks (Level 2 – 123' 11") and SFP level at the top of the fuel racks (Level 3 – 114' 11").

DCPP Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. PG&E Letter DCL-13-073 Response to Request for Additional Information Regarding Overall Integrated Plan in Response to March 12, 2012, Commission Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)
3. Procedure AR PK11-04
4. SAP documents 50808058 & 68039896 (Unit 1)
5. SAP documents 50808059 & 68039897 (Unit 2)
6. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RS2.1	**Site Area Emergency**
Lowering of spent fuel pool level to 1 ft. above top of the fuel racks (Level 3).	

Mode Applicability:

All

Definition(s):

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

Basis:

ERO Decision Making Information

This ~~IC~~ EAL addresses a significant loss of spent fuel pool inventory control ~~and makeup capability~~ leading to IMMINENT fuel damage.

For DCPP Plant SFP Level 3 is 1 ft. (plant El. 114' 11") as indicated on LI-801 (includes 1 ft. instrument uncertainly). Backup indication is also available on LI-802. The PPC point for SFP level is L0690A for both units.

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

Background

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

Post-Fukushima order EA-12-051 (ref. 1) required the installation of reliable SFP level indication capable of identifying normal minimum level (Level 1 – 134' 5"), SFP level 10 ft. above the top of the fuel racks (Level 2 – 123' 11") and SFP level at the top of the fuel racks (Level 3 – 114' 11").

DCPP Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. PG&E Letter DCL-13-073 Response to Request for Additional Information Regarding Overall Integrated Plan in Response to March 12, 2012, Commission Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number 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 1 ft. above top of the fuel racks (Level 3) for ≥ 60 minutes. (Note 1)	

Note 1: The SM/SEC/ED 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:

ERO Decision Making Information

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.

For DCPP Plant SFP Level 3 is 1 ft. (plant El. 114' 11") as indicated on LI-801 (includes 1 ft. instrument uncertainly). Backup indication is also available on LI-802. The PPC point for SFP level is L0690A for both units.

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

Background

Post-Fukushima order EA-12-051 (ref. 1) required the installation of reliable SFP level indication capable of identifying normal minimum level (Level 1 – 134' 5"), SFP level 10 ft. above the top of the fuel racks (Level 2 – 123' 11") and SFP level at the top of the fuel racks (Level 3 – 114' 11").

DCPP Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. PG&E Letter DCL-13-073 Response to Request for Additional Information Regarding Overall Integrated Plan in Response to March 12, 2012, Commission Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number 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 **EITHER** of the following areas:

Control Room (0-RM-1 or portable gamma radiation instrument)

OR

Central Alarm Station (by survey)

Mode Applicability:

All

Definition(s):

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

Basis:

ERO Decision Making Information

Areas that meet this threshold include the Control Room and the Central Alarm Station (CAS). 0-RM-1 monitors the Control Room for area radiation (ref. 1). A portable gamma radiation instrument may be installed if 0-RM-1 is out of service. The CAS is included in this EAL because of its' importance to permitting access to areas required to assure safe plant operations.

There is no permanently installed CAS area radiation monitor that may be used to assess this EAL threshold. Therefore this threshold must be assessed via local radiation survey for the CAS (ref. 1). For this EAL the Secondary Alarm Station (SAS) is not considered.

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

Background

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~~SM/SEC/ED 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.~~

DCPP Basis Reference(s):

1. FSAR Table 11.4-1 Radiation Monitors and Readouts
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(s)**
Auxiliary Building – 115' - BASTs	2, 3, 4
Auxiliary Building – 100' – BA Pumps	2, 3, 4
Auxiliary Building – 85' – Aux Control Board	2, 3, 4
Auxiliary Building – 64' – BART Tank area	2, 3, 4
Area H (below Control Room) – 100' 480V Bus area/rooms	3, 4

Mode Applicability:

2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown

Definition(s):

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

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

Basis:

<u>ERO Decision Making Information</u>

<u>The identified rooms are those where an activity must be performed to borate to cold shutdown, isolate accumulators or cooldown using RHR.</u>

<u>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.</u>

<u>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,</u>

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).

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

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.

Background

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.

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 SM/SEC/ED~~Emergency Director~~ should consider the cause of the increased radiation levels and determine if another IC may be applicable.

NOTE: EAL RA3.2 mode applicability has been limited to the applicable modes identified in Table R-2 Safe Operation & Shutdown Rooms/Areas. If due to plant operating procedure or plant configuration changes, the applicable plant modes specified in Table R-2 are changed, a corresponding change to Attachment 3 'Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases' and to EAL RA3.2 mode applicability is required."

DCCP Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Rooms/Areas Table 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.

The DCPP ISFSI is located within the OWNER CONTROLLED AREA but outside the PLANT PROTECTED AREA. Therefore SECURITY EVENTS related to the ISFSI are classified under either HU1.1 or HA1.1.

ISFSI PROTECTED AREA - Areas within the ISFSI to which access is strictly controlled in accordance with the station's Security Plan.

OWNER CONTROLLED AREA (OCA) - For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan.

PLANT PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

Category: ISFSI

Subcategory: Confinement Boundary

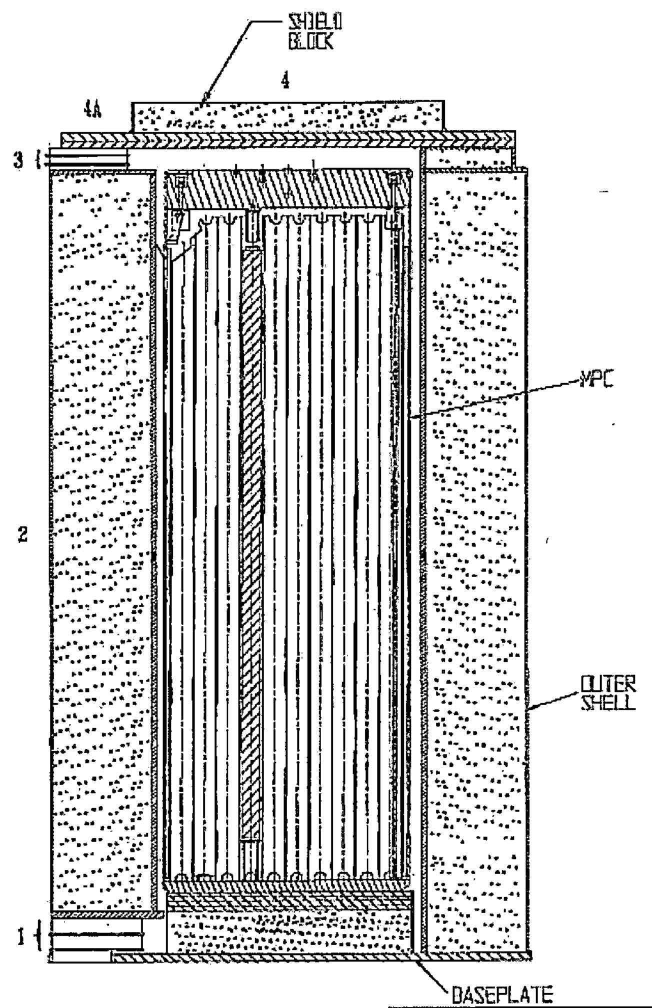
Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY.

EAL:

EU1.1 Unusual Event

Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading > Table E-1.

Table E-1 ISFSI Radiation Readings		
Dose Point Location (see figure)		**Surface Dose Rate** (mRem/hour)
1	Base vent	72
2	Mid plane	80
3	Top vent	76
4	Lid-center	22
4a	Lid-over top vents	139



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 DCPP ISFSI, the confinement boundary is defined to be the Multi-Purpose Canister (MPC).

PLANT PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

Basis:

ERO Decision Making Information

An Unusual Event is declared on the basis of the occurrence of any event of sufficient magnitude that a loaded cask confinement boundary is damaged or violated as indicated by external on-contact dose rates exceeding the maximum calculated levels of an overpack with a loaded MPC-32 canister, based on the locations in the ISFSI FSAR Figure 7.3-1 (ref. 1, 2, 3).

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.

It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

The existence of "damage" is determined by radiological survey. Exceedance of the maximum ISFSI FSAR dose rates, as noted in reference 1,~~The technical specification multiple of "2 times", which is also used in Recognition Category A IC AU1~~, 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.

The DCPP ISFSI is located wholly outside the PLANT PROTECTED AREA.

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

Background

The DCPP ISFSI Technical Specifications do not have maximum contact dose rate specified for the exterior of an overpack. The values in Table E-1 are derived from ISFSI FSAR Tables (ref 1, 2). Since the UFSAR Table 7.3-1A are the maximum calculated dose rate values, and are not expected to ever be exceeded, a conservative approach of exceeding the highest possible fuel value dose rates, plus 5 mRem/hour, was used as an indication of damage to an overpack. Note: These values are approximately 2 times the maximum expected dose rate for low burn-up fuel (ref 2).

The ISFSI includes the dry-cask storage system, the cask transfer facility, onsite transporter, and the storage pads. The dry-cask storage system is the HI-STORM 100 System. This is a canister-based storage system that stores spent nuclear fuel in a vertical orientation. It consists of three discrete components: the MPC, the HI-TRAC 125 Transfer Cask, and the HI-STORM 100 System Overpack (see pictures at end of section). The MPC provides the confinement boundary for the stored fuel. The HI-TRAC 125 Transfer Cask provides radiation shielding and structural protection of the MPC during transfer operations, while the storage overpack provides radiation shielding and structural protection of the MPC during storage. The HI-STORM 100 System is passive and does not rely on any active cooling systems to remove spent fuel decay heat. After the storage casks are placed on the storage pad, the ISFSI Technical Specifications require that the casks be inspected periodically to ensure that the air vents are not blocked. Security personnel control access to the storage area and identify and assess off-normal and emergency events. Health physics personnel perform dose rate and contamination surveys to ensure that the appropriate regulatory limits are maintained.

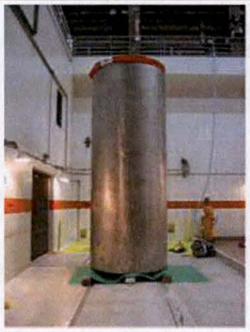
Maintenance personnel maintain the facilities including the storage casks, emergency equipment, and transport systems (ref. 4).

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.

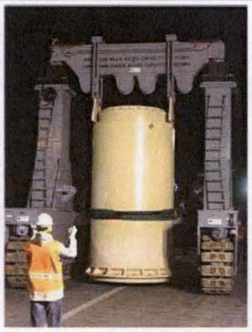
DCCP Basis Reference(s):

1. Diablo Canyon ISFSI FSAR Update, Chapter 7 Radiation Protection, Table 7.3-1A "Surface and 1 Meter Dose Rates for the Overpack with an MPC-32 69,000 MWD/MTU and 5-Year Cooling"
2. Diablo Canyon ISFSI FSAR Update, Chapter 7 Radiation Protection, Table 7.3-1B "Surface and 1 Meter Dose Rates for the Overpack with an MPC-32 32,500 MWD/MTU and 5-Year Cooling"
3. Diablo Canyon ISFSI FSAR Update, Chapter 7, Figure 7.3-1 "Cross Section Elevation of the Generic Hi-Storm 100S Overpack with Dose Point Locations."
4. NRC Materials License No. SNM-2511, LICENSE FOR INDEPENDENT STORAGE OF SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE, Safety Evaluation Report
5. NEI 99-01 E-HU1

DCPP ISFSI HI-STORM 100 System



MPC (CONFINEMENT BOUNDARY)



Transporter with Hi-TRAC Transfer Cask

HI-STORM Storage Casks (Overpack)

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 plant electrical power can compromise plant SAFETY SYSTEM operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of power to or degraded voltage on the 125V DC vital buses.

5. Loss of Communications

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

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.

DEGRADED PERFORMANCE - As applied to hazardous event thresholds, damage significant enough to cause concern regarding the operability or reliability of the affected safety system train.

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.

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 RCS inventory results in RCS water level less than a procedurally designated lower limit for ≥ 15 minutes. (Note 1)	

Note 1: The SM/SEC/ED 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):

REDUCED INVENTORY CONDITION (RIC) - The condition existing whenever RCS water level is lower than 3 feet below the reactor vessel flange (below 111-foot elevation) with fuel in the core.

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:

ERO Decision Making Information

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.

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.

Background

With the plant in Cold Shutdown, RCS water level is normally maintained above 25% Cold Calibration Pressurizer level (~129 ft. elevation). However, if RCS level is being controlled below 25%, or if level is being maintained in a procedurally 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 (ref. 2).

With the plant in Refueling mode, RCS water level is normally maintained at or above the reactor vessel flange (Technical Specifications requires at least 23 ft. of water above the top of the reactor vessel flange in the refueling cavity during refueling operations) (ref. 1). However, RCS level may be maintained below the reactor vessel flange if in "lowered inventory" or "REDUCED INVENTORY" condition (ref. 2).

This ~~IC~~ EAL addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a procedurally specified 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~~ This condition~~s~~ is considered to be a potential degradation of the level of safety of the plant.

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.

~~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*]).~~

DCPP Basis Reference(s):

1. Technical Specification 3.9.7, Refueling Cavity Water Level
2. OP A-2: II, U1 Reactor Vessel - Draining the RCS to the Vessel Flange - With Fuel in Vessel
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 Structure Sumps
• Reactor Cavity Sump
• PRT
• RCDT
• CCW Surge Tank(s)
• Auxiliary Building Sump
• RWST
• RHR Room Sumps (alarm **only**)
• MEDT

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

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).

RCS LEAKAGE – RCS leakage shall be:

a. Identified Leakage

1. Leakage, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leak off), that is captured and conducted to collection systems or a sump or collecting tank;
2. 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;

3. Reactor Coolant System (RCS) leakage through a steam generator to the secondary system (primary to secondary leakage).

b. Unidentified Leakage

All leakage (except RCP seal water injection or leak off) that is not identified leakage.

c. Pressure Boundary Leakage

Leakage (except primary to secondary leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

d. 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).

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

NOTE: If an isolation valve could not be accessed due to local conditions (i.e. high radiation, temperature, etc.) then that would also make a leak unisolable, even though the inaccessible valve could isolate the leak.

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:

ERO Decision Making Information

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. The ability to monitor RCS level includes level instrumentation as well as direct and indirect (e. g. camera) visual observation.

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.

Background

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

In this EAL, the ability to monitor RCS level is lost such that RCS inventory loss must be detected by indirect leakage indications. The ability to monitor RCS level includes level instrumentation as well as direct and indirect (e. g. camera) visual observation. 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 to maintain RCS inventory, 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).

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~~ RCS LEAKAGE. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

~~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.~~

DCPP Basis Reference(s):

1. OP AP SD-2, "Loss of RCS Inventory

2. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory.

EAL:

CA1.1	**Alert**
Loss of RCS inventory as indicated by reactor vessel level < 107 ft. 6 in. (107.5 ft.) on RVRLIS, LI-400 standpipe or ultrasonic sensor.	
OR	
< 67.5% RVLIS full range (RVLIS equivalent to 107 ft. 6 in.).	

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

None

Basis:

ERO Decision Making Information

When reactor vessel water level decreases to 107 ft. 6 in. el., RCS level is ~21 in. above the bottom of the RCS hot leg penetration. This is the minimum procedurally allowed RCS level to preclude vortexing of the RHR pumps while in Shutdown Cooling. This level can be monitored by:

- RVRLIS
- LI-400 standpipe
- Ultrasonic sensor

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~~ Decay 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 ~~#1~~, a lowering of RCS water level below the specified level ~~(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.

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.

Background

The purpose of the Reactor Vessel Refueling Level Instrumentation System (RVRLIS) is to provide reactor vessel and refueling cavity level indication during refueling, when the vessel head will be removed, and during drainage to half loop. The system is designed to be used only when the RCS is at near atmospheric pressure or when a vacuum is being established for refill operations. The wide range and narrow range RVRLIS (if required) and the LI-400 standpipe systems remain in service from the time RCS level is lowered below 25% Cold Calibrated Pressurizer level until just prior to pressurizing the RCS. Narrow Range RVRLIS is required if reduced inventory conditions (below 111 ft. elevation) are planned.

The LI-400 standpipe is a magnetic level indicator (LI-400A, B, C standpipe) and provides local indication of reactor vessel refueling level. The indicator is mounted on the outside of the secondary shield wall (crane wall) and can be viewed from the 91 ft. elevation of Containment. The indicator is composed of three mechanical flag indicator units.

RVRLIS, LI-400 standpipe and ultrasonic detectors are off-scale low (105 ft. 9 in.) when reactor vessel water level drops below the elevation of the bottom of the RCS hot leg penetration. The ultrasonic sensor is installed during an outage and measures level on one of the hot legs.

The purpose of the Reactor Vessel Level Instrumentation System (RVLIS) is to measure the level of the water or the relative void content of the coolant in the reactor vessel. The RVLIS setpoint corresponding to the minimum RHR pump operation limit was obtained as follows (ref. 2, 3, 4):

- Full range:
  - Per SC-I-87B, span = 494.9 in. and 0% = 79.6536 feet
  - % span/in. = 100 / 494.9 = 0.20206%/in. and minimum RCS level for RHR operation (from above) = 107.5 feet
  - (107.5 - 79.6536) x 12 x 0.20206 = 67.5%

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 #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~~

DCPP Basis Reference(s):

1. Plant Drawing No. 57729
2. OP A-2:III, Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in Vessel Drain-Down
3. Instrument Scaling Calculation SC-I-87B, Reactor Vessel Level Instrumentation System and Subcooled Margin Monitor
4. OP AP SD-0 Loss of, or Inadequate Decay Heat Removal

5. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory.

EAL:

CA1.2 Alert

RCS water level **cannot** be monitored for ≥ 15 minutes. (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 SM/SEC/ED should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-1 Sumps / Tanks
• Containment Structure Sumps • Reactor Cavity Sump • PRT • RCDT • CCW Surge Tank(s) • Auxiliary Building Sump • RWST • RHR Room Sumps (alarm **only**) • MEDT

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

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).

RCS LEAKAGE – RCS leakage shall be:

a. Identified Leakage

   1. Leakage, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leak off), that is captured and conducted to collection systems or a sump or collecting tank;

2. 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;
3. Reactor Coolant System (RCS) leakage through a steam generator to the secondary system (primary to secondary leakage).

b. Unidentified Leakage

All leakage (except RCP seal water injection or leak off) that is not identified leakage.

c. Pressure Boundary Leakage

Leakage (except primary to secondary leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

e. 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).

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

NOTE: If an isolation valve could not be accessed due to local conditions (i.e. high radiation, temperature, etc.) then that would also make a leak unisolable, even though the inaccessible valve could isolate the leak.

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:

ERO Decision Making Information

In this EAL, the ability to monitor RCS level would be unavailable for greater than 15 minutes, and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). The ability to monitor RCS level includes level instrumentation as well as direct and indirect (e. g. camera) visual observation.

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).

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.

Background

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 by direct or indirect methods, 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])~~.

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.

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.~~

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

DCPP Basis Reference(s):

1. OP AP SD-2, Loss of RCS Inventory
2. OP AP-1, Excessive Reactor Coolant System Leakage
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**
With CONTAINMENT CLOSURE **not** established, RVLIS full range < 62.1%. (Note 12)	

Note 12: With RVLIS out-of-service, classification shall be based on CS1.3 or CG1.2 if RCS inventory **cannot** be monitored.

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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 Containment Closure.

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:

ERO Decision Making Information

This EAL is only applicable when RVLIS is in service (see Note 12).

When reactor vessel water level lowers to 62.1%, water level is six inches below the elevation of the bottom of the RCS hot leg penetration.

Six inches below the elevation of the bottom of the RCS hot leg penetration can be monitored only by RVLIS.

Other reactor vessel water level monitoring systems (e.g., RVRLIS, LI-400 standpipe, ultrasonic sensor, RVLIS upper range) are downscale-low when water level drops below the elevation of the bottom of the RCS hot leg penetration.

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

Background

When reactor vessel water level drops significantly below the elevation of the bottom of the RCS hot leg penetration, all sources of RCS injection have failed or are incapable of making up for the inventory loss.

The RVLIS setpoint corresponding to six inches below the elevation of the bottom of the RCS hot leg penetration was obtained as follows (ref. 1, 2, 3, 4):

- Per SC-I-87B, span = 494.9 in. and 0% = 79.6536 feet

- % span/in. = 100 / 494.9 = 0.20206%/in. and bottom of the hot leg (from above) = 105.75 feet
- (105.75 - 6 - 79.6536) x 12 x 0.20206 = 62.1%

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

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).

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~~ RCS level cannot be restored, fuel damage is probable.

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS/reactor vessel levels of EALs ~~1.b~~CS1.1 and ~~2.b~~CS2.2 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, 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.

DCPP Basis Reference(s):

1. Plant Drawing No. 57729
2. OP A-2:III, Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in Vessel
3. AP E-55, Equipment Elevations for RCS Flood-Up and Drain-Down
4. Instrument scaling calculation SC-I-87B, "Reactor Vessel Level Instrumentation System and Subcooled Margin Monitor
5. AD8.DC54, Containment Closure
6. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.2	**Site Area Emergency**
With CONTAINMENT CLOSURE established, RVLIS full range < 56.6% (Top of Fuel). (Note 12)	

Note 12: With RVLIS out-of-service, classification shall be based on CS1.3 or CG1.2 if RCS inventory **cannot** be monitored.

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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 Containment Closure.

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.

Bases:

ERO Decision Making Information

This EAL is only applicable when RVLIS is in service (see Note 12).

When reactor vessel water level drops below RVLIS full range indication of 56.6% core uncovery is about to occur. This level drop can only be remotely monitored by reactor vessel Level Instrumentation System (RVLIS).

Other reactor vessel water level monitoring systems (e.g., RVRLIS, LI-400 standpipe, ultrasonic sensor, RVLIS upper range) are downscale-low when water level drops below the elevation of the bottom of the RCS hot leg penetration.

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

Background

The RVLIS setpoint corresponding to the top of fuel was obtained as follows (ref. 1, 2, 3, 4):

- Per SC-I-87B, span = 494.9 in. and 0% = 79.6536 feet
- % span/in. = 100 / 494.9 = 0.20206%/in. and top of core = 103 feet
- (103 - 79.6536) x 12 x 0.20206 = 56.6%

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

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).

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

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

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

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

~~The inability to monitor~~ 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]).~~

~~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.

DCPP Basis Reference(s):

1. Plant Drawing No. 57729
2. OP A-2:III, Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in Vessel
3. AP E-55, Equipment Elevations for RCS Flood-Up and Drain-Down

4. Instrument scaling calculation SC-I-87B, “Reactor Vessel Level Instrumentation System and Subcooled Margin Monitor
5. AD8.DC54, Containment Closure
6. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.3	**Site Area Emergency**

RCS water level **cannot** be monitored for ≥ 30 minutes. (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 uncover.
- **Any** Bridge (Manipulator) Crane Radiation Monitor > 9 R/hr.
- Erratic Source Range Monitor indication.

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

Table C-1 Sumps / Tanks
• Containment Structure Sumps
• Reactor Cavity Sump
• PRT
• RCDT
• CCW Surge Tank(s)
• Auxiliary Building Sump
• RWST
• RHR Room Sumps (alarm **only**)
• MEDT

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

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

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).

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:

ERO Decision Making Information

In this EAL, the ability to monitor RCS level would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). The ability to monitor RCS level includes level instrumentation as well as direct and indirect (e. g. camera) visual observation.

The reactor vessel inventory loss may be detected by the radiation monitors or erratic source range monitor indication. As water level in the reactor vessel lowers, the dose rate above the core will rise. The Bridge Crane Radiation Monitors can be monitored either locally, or remotely on the Viewpoint software.

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.1).

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 RCS ~~(reactor vessel/RCS [PWR] or RPV [BWR])~~.

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

Background

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.

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.

The dose rate due to this core shine should result in increased Bridge (Manipulator) Crane Radiation Monitor indication. A reading of 9 R/hr (90% of instrument scale) is indicative of core uncovery. There are a number of variables governing the projected dose rate from an actual core uncover (ref. 2).

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.

~~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.

DCPP Basis Reference(s):

1. Nuclear Safety Analysis Center (NSAC), 1980, "Analysis of Three Mile Island - Unit 2 Accident," NSAC-1
2. EP-EALCALC-DCPP-1603 Radiation Monitor Readings for Core Uncovery During Refueling
3. 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

RVLIS full range < 56.6% (Top of Fuel) for ≥ 30 minutes. (Notes 1, 12)

AND

Any Containment Challenge indication, Table C-2.

Note 1: The SM/SEC/ED 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.

Note 12: With RVLIS out-of-service, classification shall be based on CS1.3 or CG1.2 if RCS inventory **cannot** be monitored.

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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 Containment Closure.

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>ERO Decision Making Information</u>

<u>This EAL is only applicable when RVLIS is in service (see Note 12).</u>

When reactor vessel water level drops below RVLIS full range indication of 56.6% core uncovery is about to occur. This level drop can only be remotely monitored by Reactor Vessel Level Instrumentation System (RVLIS).

Other reactor vessel water level monitoring systems (e.g., RVRLIS, LI-400 standpipe, ultrasonic sensor, RVLIS upper range) are downscale-low when water level drops below the elevation of the bottom of the RCS hot leg penetration.

Three conditions are associated with a challenge to Containment:

1. CONTAINMENT COSURE not established
2. Containment hydrogen ≥ 4%
3. UNPLANNED rise in Containment pressure (Containment pressure changes due to ventilation system changes do not constitute a containment challenge)

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.

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*] o r 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*])~~.

Background

The RVLIS setpoint corresponding to the top of fuel was obtained as follows (ref. 1, 2, 3, 4):

- Per SC-I-87B, span = 494.9 in. and 0% = 79.6536 feet
- % span/in. = 100 / 494.9 = 0.20206%/in. and top of core = 103 feet
- (103 - 79.6536) x 12 x 0.20206 = 56.6%

Three conditions are associated with a challenge to Containment:

1. CONTAINMENT COSURE 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 be required.

2. Containment hydrogen ≥ 4% - The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations. To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers are likely to have occurred. Operation of the Containment Hydrogen Recombiner with Containment hydrogen concentrations above 4.0% could result in ignition of the hydrogen. If in operation, containment hydrogen can be monitored in the Control Room on ANR-82/ANR-83 and PAM1 following local equipment initialization (ref. 6, 7)
3. UNPLANNED rise in Containment pressure - In the operating modes associated with this EAL, Containment pressure is expected to remain very low; thus, an elevated Containment pressure resulting from an UNPLANNED rise above near-atmospheric pressure conditions may be indicative of a challenge to the Containment barrier. Containment pressure changes due to ventilation system changes do not constitute a containment challenge.

Under the conditions specified by this EAL, continued lowering of reactor vessel water level is indicative of a loss of inventory control with a challenge to the Containment. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the reactor vessel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS or reactor vessel water level drop and potential core uncovery. The inability to restore and maintain level inventory within 30 minutes after reaching this condition in combination with a Containment challenge infers a failure of the RCS barrier, Loss of the Fuel Clad barrier and a Potential Loss of Containment.

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.

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.

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*.

DCPP Basis Reference(s):

1. Plant Drawing No. 57729
2. OP A-2:III, Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in Vessel
3. AP E-55, Equipment Elevations for RCS Flood-Up and Drain-Down
4. Instrument scaling calculation SC-I-87B, "Reactor Vessel Level Instrumentation System and Subcooled Margin Monitor
5. AD8.DC54, Containment Closure
6. CA-3, Hydrogen Flammability in Containment
7. OP H-9, INSIDE CONT H2 RECOMB SYSTEM
8. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CG1.2 General Emergency

RCS level **cannot** be monitored for ≥ 30 minutes. (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 uncover.
- **Any** Bridge (Manipulator) Crane Radiation Monitor > 9 R/hr.
- Erratic Source Range Monitor indication.

AND

Any Containment Challenge indication, Table C-2.

Note 1: The SM/SEC/ED 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 Structure Sumps
• Reactor Cavity Sump
• PRT
• RCDT
• CCW Surge Tank(s)
• Auxiliary Building Sump
• RWST
• RHR Room Sumps (alarm **only**)
• MEDT

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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 Containment Closure.

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).

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:

ERO Decision Making Information

In this EAL, the ability to monitor RCS level would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). The ability to monitor RCS level includes level instrumentation as well as direct and indirect (e. g. camera) visual observation.

The reactor vessel inventory loss may be detected by the radiation monitors or erratic source range monitor indication. As water level in the reactor vessel lowers, the dose rate above the core will rise. The Bridge Crane Radiation Monitors can be monitored either locally, or remotely on the Viewpoint software.

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. 1).

Source Range indication can be seen on Source Range Detectors NI-31 & 32 as well as the Gammametrics detectors.

Three conditions are associated with a challenge to Containment:

1. CONTAINMENT COSURE not established

2. Containment hydrogen ≥ 4%
3. UNPLANNED rise in Containment pressure (Containment pressure changes due to ventilation system changes do not constitute a containment challenge)

During periods when installed containment hydrogen gas monitors are out-of-service, 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.

Background

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.

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.

The dose rate due to this core shine should result in increased Bridge (Manipulator) Crane Radiation Monitor indication. A reading of 9 R/hr (90% of instrument scale) is indicative of core uncovery. There are a number of variables governing the projected dose rate from an actual core uncover (ref. 5).

Three conditions are associated with a challenge to Containment:

1. CONTAINMENT COSURE 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.2). If containment closure is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to GE would not be required.
2. Containment hydrogen ≥ 4% - The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations. To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers are likely to have occurred. Operation of the Containment Hydrogen Recombiner with Containment hydrogen concentrations above 4.0% could result in ignition of the hydrogen. Containment hydrogen can be monitored in the Control Room on ANR-82/ANR-83 and PAM1 following local equipment initialization (ref. 3, 4)

3. ~~UNPLANNED rise in Containment pressure - In the operating modes associated with this EAL, Containment pressure is expected to remain very low; thus, an elevated Containment pressure resulting from an UNPLANNED rise above near-atmospheric pressure conditions may be indicative of a challenge to the Containment barrier. Containment pressure changes due to ventilation system changes do not constitute a containment challenge.~~

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*]).~~

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

DCPP Basis Reference(s):

1. Nuclear Safety Analysis Center (NSAC), 1980, "Analysis of Three Mile Island - Unit 2 Accident," NSAC-1
2. AD8.DC54, Containment Closure
3. OP H-9, INSIDE CONT H2 RECOMB SYSTEM
4. CA-3, Hydrogen Flammability in Containment
5. EP-EALCALC-DCPP-1603 Radiation Monitor Readings for Core Uncovery During Refueling
6. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Vital AC Power

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

EAL:

CU2.1 Unusual Event

AC power capability, Table C-3, to Unit 1 or Unit 2 vital 4.16KV buses F, G and H reduced to a single power source for ≥ 15 minutes. (Note 1)

AND

A failure of that single power source will result in loss of **all** AC power to SAFETY SYSTEMS.

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

Table C-3 AC Power Capability

	Unit 1	Unit 2
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

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:

ERO Decision Making Information

For emergency classification purposes, "capability" means that, whether or not the buses are actually powered from it, an AC power source(s) can be aligned to the vital buses within 15 minutes:

- By a clear procedure path,
  and
- Breakers and equipment are readily available to power up the bus within the allotted time frame.

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.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to ~~an essential~~ vital 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~~ vital 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~~ vital buses being back-fed from an offsite power source.
- If the 1-1- or 2-1 transformer is unavailable, the other unit's #1 transformer (2-1 or 1-1) can be used to supply the startup bus through the startup bus cross tie breaker.

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

When filling out the ENF form, this event can be Unit 1, Unit 2 or Unit 1 and 2.

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

Background

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 vital buses.

4.16KV buses F, G and H are the emergency (vital) buses. Each bus has three sources of power (see figure below).

One offsite source is from the 230KV switchyard via 12KV startup transformer 1-1 (2-1) to the 4.16KV startup transformer 1-2 (2-2).

Another method to obtain offsite power is by back feeding the vital buses from the 500KV switchyard through the main transformer to the 4.16KV unit auxiliary transformer 1-2 (2-2). This is normally only done post-trip when 500KV power is available.

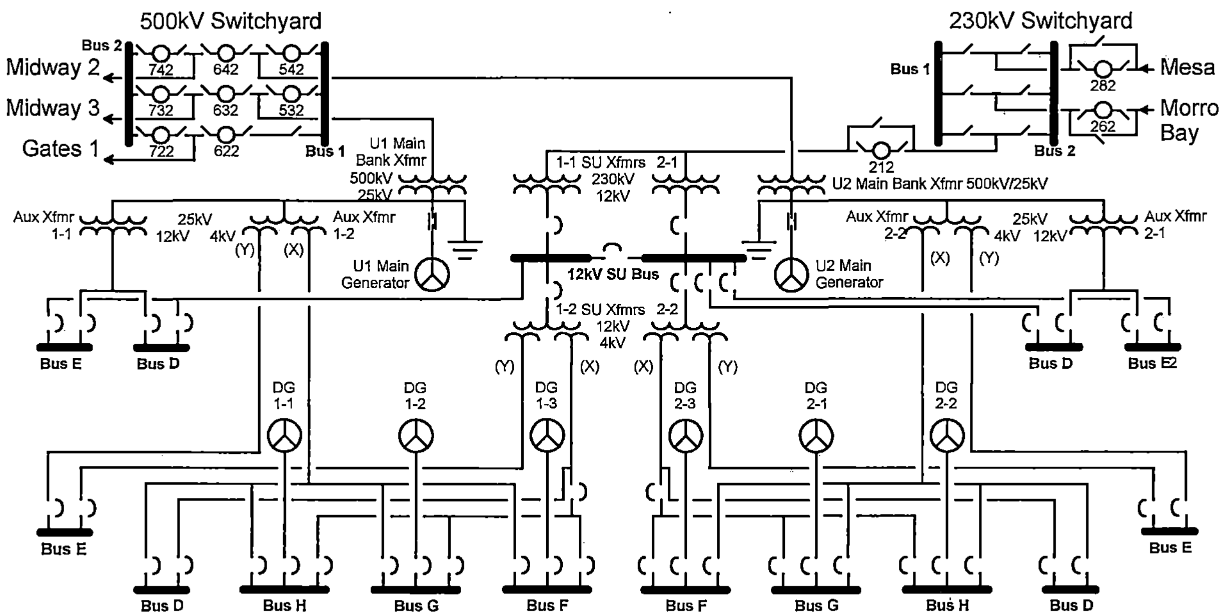
In addition, each units vital buses F, G and H have an onsite emergency diesel generator which can supply electrical power to its associated bus automatically in the event that the preferred source becomes unavailable (ref. 1-6).

Refer to CP M-10 Fire Protection of Safe Shutdown Equipment for a list of SAFETY SYSTEMS.

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.

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

DCPP Electrical Distribution System



DCPP Basis Reference(s):

1. UFSAR, Section 8.2.2
2. UFSAR, Section 8.3.1.6
3. OP AP SD-1, Loss of AC Power
4. OP AP-2, Loss of Offsite Power
5. OP J-2:V, Backfeeding the Unit From the 500kV System
6. ECA-0.0, Loss of All Vital AC Power
7. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Vital AC Power

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

EAL:

CA2.1 Alert

Loss of **all** offsite and **all** onsite AC power capability, Table C-3, to Unit 1 or Unit 2 vital 4.16KV buses F, G and H for ≥ 15 minutes. (Note 1)

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

Table C-3 AC Power Capability

	Unit 1	Unit 2
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

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:

ERO Decision Making Information

For emergency classification purposes, "capability" means that, whether or not the buses are actually powered from it, an AC power source(s) can be aligned to the vital buses within 15 minutes:

- By a clear procedure path,
  and
- Breakers and equipment are readily available to power up the bus within the allotted time frame.

This ~~IC~~ EAL addresses a total loss of AC power for greater than 15 minutes 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.

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

When filling out the ENF form, this event can be Unit 1, Unit 2 or Unit 1 and 2.

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

Background

The vital 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant (see figure below).

Unit 1(2) 4.16KV buses F, G and H are the emergency (vital) buses. Each bus has three sources of power.

One offsite source is from the 230KV switchyard via 12KV startup transformer 1-1 (2-1) to the 4.16KV startup transformer 1-2 (2-2).

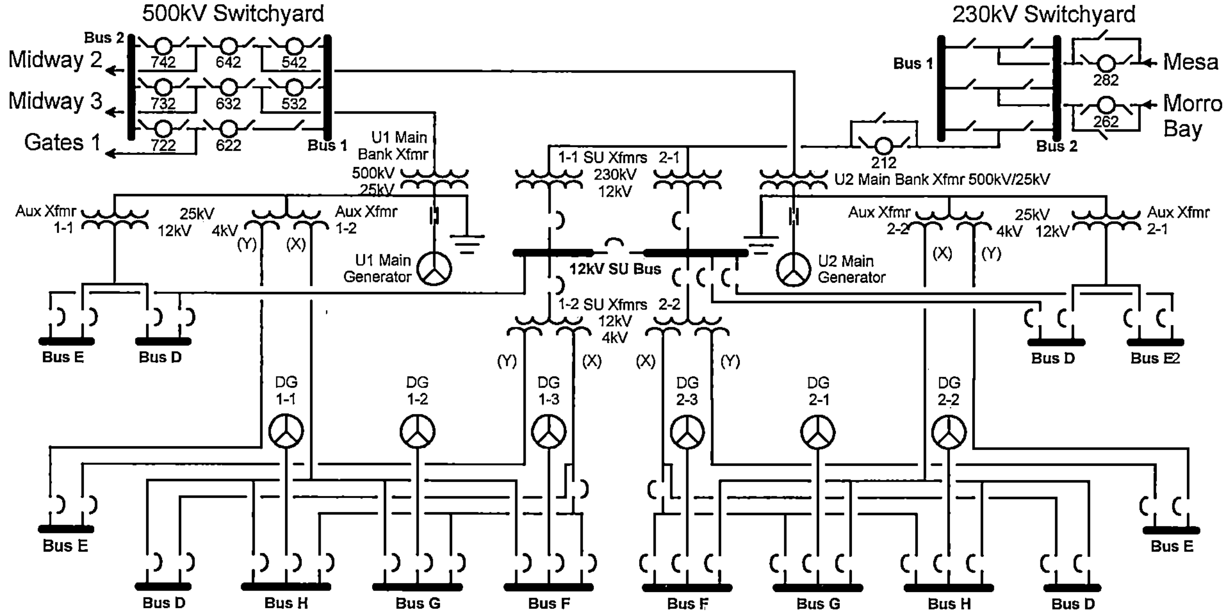
Another method to obtain offsite power is by back feeding the vital buses from the 500KV switchyard through the main transformer to the 4.16KV unit auxiliary transformer 1-2 (2-2). This is normally only done post-trip when 500KV power is available.

In addition, each units vital buses F, G and H have an onsite emergency diesel generator which can supply electrical power to its associated bus automatically in the event that the preferred source becomes unavailable (ref. 1-6).

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~~ vital 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.

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

DCPP Electrical Distribution System



DCPP Basis Reference(s):

1. UFSAR, Section 8.2.2
2. UFSAR, Section 8.3.1.6
3. OP AP SD 1, Loss of AC Power
4. OP AP-2, Loss of Offsite Power
5. OP J-2:V, Backfeeding the Unit From the 500kV System
6. ECA-0.0, Loss of All Vital AC Power
7. 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.

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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 "Containment Closure."

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).

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:

ERO Decision Making Information

In the absence of reliable RCS temperature indication caused by a loss of decay heat removal capability, classification should be based on EAL CU3.2 should RCS level indication be subsequently lost.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit of 200°F when the heat removal function is available does not warrant a classification.

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

Background

Numerous instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1). These may include but are not limited to (ref. 2):

- TR413 Loop 1 Wide Range Temperature
- TR423 Loop 2 Wide Range Temperature

- TR433 Loop 3 Wide Range Temperature
- TR443 Loop 4 Wide Range Temperature
- WR T_{hot} recorders 0-700°F
- WR T_{cold} recorders 0-700°F
- CETs
- RHR System temperatures (when RHR is in service)

This IC addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit~~, or the inabili ty 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 SM/SEC/ED~~Emergency Director~~ should also refer to IC CA3.

~~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.~~

DCPP Basis Reference(s):

1. DCPP Technical Specifications Table 1.1-1 Modes
2. OP L-1, Plant Heatup From Cold Shutdown to Hot Standby
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 **all** RCS level indication for ≥ 15 minutes. (Note 1)	

Note 1: The SM/SEC/ED 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):

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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 "Containment Closure."

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).

Basis:

ERO Decision Making Information

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 SM/SEC/ED~~Emergency Director~~ should also refer to IC CA3.

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.

Background

Reactor vessel water level is normally monitored using the following instruments (ref. 1):

- RVRLIS
- LI-400 Standpipe
- RVLIS

- Ultrasonic level detectors

AP E-55, "Equipment Elevations for RCS Flood-Up and Drain-Down", provides a cross-reference of indicated water levels and key plant elevations

Numerous instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 2). These may include but are not limited to (ref. 3):

- TR413 Loop 1 Wide Range Temperature
- TR423 Loop 2 Wide Range Temperature
- TR433 Loop 3 Wide Range Temperature
- TR443 Loop 4 Wide Range Temperature
- WR T_{hot} recorders 0-700°F
- WR T_{cold} recorders 0-700°F
- CETs
- RHR System temperatures (when RHR is in service)

~~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.

DCPP Basis Reference(s):

1. AP E-55, "Equipment Elevations for RCS Flood-Up and Drain-Down
2. DCPP Technical Specifications Table 1.1-1
3. OP L-1, Plant Heatup From Cold Shutdown to Hot Standby
4. 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 does **not** apply during water-solid plant conditions).

Note 1: The SM/SEC/ED 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.

<table>
<tr><th colspan="3">Table C-4: RCS Heat-up Duration Thresholds</th></tr>
<tr><th>RCS Status</th><th>CONTAINMENT CLOSURE Status</th><th>Heat-up Duration</th></tr>
<tr><td>INTACT (but not REDUCED INVENTORY)</td><td>N/A</td><td>60 minutes*</td></tr>
<tr><td rowspan="2">Not INTACT
OR
REDUCED INVENTORY</td><td>established</td><td>20 minutes*</td></tr>
<tr><td>not established</td><td>0 minutes</td></tr>
<tr><td colspan="3">* If an RCS heat removal system is in operation within this time frame and RCS temperature is trending down, the EAL is not applicable.</td></tr>
</table>

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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 Containment Closure.

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).

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.

REDUCED INVENTORY - The condition existing whenever RCS water level is lower than 3 feet below the reactor vessel flange (below 111-foot elevation) with fuel in the core.

Basis:

ERO Decision Making Information

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 the RCS is INTACT in Mode 5 or based on time to boil data when in Mode 6 or the RCS is not INTACT in Mode 5.

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.

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

Background

Numerous instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1). These may include but are not limited to (ref. 2):

- TR413 Loop 1 Wide Range Temperature
- TR423 Loop 2 Wide Range Temperature
- TR433 Loop 3 Wide Range Temperature
- TR443 Loop 4 Wide Range Temperature
- WR T_{hot} recorders 0-700°F
- WR T_{cold} recorders 0-700°F
- CETs
- RHR System temperatures (when RHR is in service)

PI-403A, PI-405 and PI-405A display on VB2, with digital values available on PPC, SPDS and SCMM. Digital readouts can display changes of less than 10 psig.

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.

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 in the absence of RCS temperature monitoring capability.

DCPP Basis Reference(s):

1. DCPP Technical Specifications Table 1.1-1
2. OP L-1, Plant Heatup From Cold Shutdown to Hot Standby
3. 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**
< 105 VDC bus voltage indications on Technical Specification **required** 125 VDC vital buses for ≥ 15 minutes. (Note 1)	

Note 1: The SM/SEC/ED 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):

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

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

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

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

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

Basis:

<u>ERO Decision Making Information</u>

<u>Minimum battery voltage of 105 VDC is the voltage below which supplied loads may not function (ref. 2, 3, 4).</u>

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~~<u>R</u>.

<u>Background</u>

<u>The vital 125 VDC system consists of three independent networks per unit. Each network contains the following components:</u>

- Battery
- Battery charger
- Standby battery charger to allow maintenance and/or testing
- Distribution panels
- Ground detector

The vital 125 VDC batteries and distribution panels are located in Area "H" of the 115 feet elevation of the Auxiliary Building. There are a total of three batteries per unit, 11(21), 12(22), and 13(23). The batteries are sized to provide sufficient power to operate the associated DC loads for the time necessary to safely shut down the unit, should a 480-VAC source to one or more battery chargers be unavailable (ref. 1, 2, 3).

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.

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

DCPP Basis Reference(s):

1. UFSAR, Section 8.3.2.2.2
2. OP AP-23, Loss of Vital DC Bus
3. ECA-0.0, Loss of All Vital AC Power
4. Notification 50804190 DC Bus Voltage Trigger for EALs
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 offsite communication methods.

OR

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

Table C-5 Communication Methods			
System	**Onsite**	**Offsite**	**NRC**
Unit 1, Unit 2 and TSC Radio Consoles	X	X	
DCPP Telephone System (PBX)	X	X	X
Portable radio equipment (handie-talkies)	X		
Operations Radio System	X	X	
Security Radio Systems	X		
CAS and SAS Consoles	X	X	X
Fire Radio System	X		
Hot Shutdown Panel Radio Consoles	X	X	X
Public Address System	X		
NRC FTS			X
Mobile radios	X		
Satellite phones	X	X	X
Direct line (ATL) to the County and State OES		X	

Mode Applicability:

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

Definition(s):

None

Basis:

ERO Decision Making Information

Onsite, offsite and NRC communications include one or more of the systems listed in Table C-5 (ref. 1, 2, 3).

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 Offsite Response Organizations (OROs) and the NRC.

Background

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 county EOCs~~.~~

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

DCPP Basis Reference(s):

1. UFSAR, Section 9.5.2
2. Emergency Plan Section 7.2 Communications Equipment
3. AR PK15-23, Communications
4. 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 • Tsunami • Other events with similar hazard characteristics as determined by the SM/SEC/ED

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

DEGRADED PERFORMANCE – As applied to hazardous event thresholds, event damage significant enough to cause concern regarding the operability or reliability of the affected safety system train.

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.

TORNADO - A violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

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:

ERO Decision Making Information

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.

The indications of DEGRADED PERFORMANCE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

In modes 5, 6 and defueled, the appropriate plant configuration based Outage Safety Checklist in AD8.DC55 "Outage Safety Scheduling" should be consulted to identify required equipment supporting each of the specified safety functions (ref. 1).

With respect to event damage caused by an equipment failure resulting in a FIRE or EXPLOSION, no emergency classification is required in response to a FIRE or EXPLOSION resulting from an equipment failure if the only safety system equipment affected by the event is that upon which the failure occurred. An emergency classification is required if a FIRE or EXPLOSION caused by an equipment failure damages safety system equipment that was otherwise functional or operable (i.e., equipment that was not the source/location of the failure). For example, if a FIRE or EXPLOSION resulting from the failure of a piece of safety system equipment causes damage to the other train of the affected safety system or another safety system, then an emergency declaration is required in accordance with this IC and EAL.

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

Background

This condition represents an actual or potential substantial degradation of the level of safety of the plant. Due to this actual or potential substantial degradation, this condition can significantly reduce~~s~~ 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.

~~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.

DCPP Basis Reference(s):

1. AD8.DC55 Outage Safety Scheduling
2. 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 PLANT PROTECTED AREA, bomb threats, sabotage attempts, and actual security compromises threatening loss of physical control of the plant.

2. Seismic Event

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

3. Natural or 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 ISFSI or PLANT PROTECTED AREA or which may affect operability of equipment needed for safe shutdown

5. Control Room Evacuation

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

6. SM/SEC/ED Judgment

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

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):

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

OWNER CONTROLLED AREA (OCA) - For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan.

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.

SECURITY EVENT - Any incident representing an attempted, threatened, or actual breach of the security system or reduction of the operational effectiveness of that system. A security event can result in either a SECURITY CONDITION or HOSTILE ACTION.

Basis:

ERO Decision Making Information

The intent of the EAL is to ensure that appropriate notifications for the security threat are made in a timely manner. The DCPP Security and Safeguards Contingency Plan provides a description of SECURITY EVENTS indicative of a potential loss of the level of safety of the plant. Events at the Unusual Event level include credible threats to attack or use a bomb against the plant, or involve extortion, coercion or HOSTAGE threats.

NOTE: **DO NOT** revise this Technical Basis Document to add any identifying information to any SECURITY EVENT codes, and do not remove this note.

Threats under this EAL include the following SECURITY EVENT categories:

- SE-1, SE-2, SE-3, SE-7, SE-9, SE-10, SE-11, SE-12, SE-13, SE-14, SE-15, SE-16, SE-17, SE-18, SE-19, SE-20 & SE-21

Security Watch Commanders are the designated on-site 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 DCPP Security and Safeguards Contingency Plan (Safeguards) information. (ref. 1).

~~EAL #1~~The first threshold:

The Security Watch Commanders, as the trained individuals~~references (site-specific security shift supervision because these are the individuals trained to~~ confirm that a SECURITY EVENT is occurring or has occurred, and whether or not the event **is** or **is not** a HOSTILE ACTION. 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 receipt of a ~~credibil~~credible~~ity of the~~security threat is assessed in accordance with ~~(site-specific procedure)~~the Security and Safeguards Contingency Plan (ref. 1). This EAL is met when the plant receives information from the NRC or other reliable source, such as the FBI.

~~EAL #3~~The third threshold:

~~addresses the threat from the impact of an aircraft on the plant.~~ This EAL is met when the plant receives information regarding an aircraft threat from the NRC or other reliable source, such as the FBI, FAA, or NORAD, and the aircraft is more than 30 minutes away from the plant. In this EAL the threat from the impact of an aircraft on the plant is assessed. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with the Security and Safeguards Contingency Plan~~(site-specific procedure)~~.

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

Background

The security shift supervision is defined as the Security Watch Commander.

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.

Threat information may come from various sources, including the NRC or FBI. Only the plant to which the specific threat is made need declare the Unusual Event.

This EAL is based on the DCPP Security and Safeguards Contingency 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, as outlined in DCPP Administrative Procedure XI1.ID2 (ref. 3).

SECURITY EVENTS assessed as HOSTILE ACTIONS are classifiable under ICs HA1~~,~~ and HS1 ~~and HG1~~.

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~~]~~*.

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 and Safeguards Contingency Plan.

DCPP Basis Reference(s):

1. DCPP Security and Safeguards Contingency Plan
2. DCPP Procedures (Procedure names and designations are controlled due to the nature of Safeguards and 10 CFR § 2.39 information.)
3. DCPP Administrative Procedure XI1.ID2 "Regulatory Reporting Requirements and Reporting Process"
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HA1.1	**Alert**

A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Watch Commander.

OR

A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.

Mode Applicability:

All

Definition(s):

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

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

OWNER CONTROLLED AREA (OCA) - For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan.

PLANT PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

ERO Decision Making Information~~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.~~

The intent of the EAL is to ensure that appropriate notifications are made in a timely manner. The DCPP Safeguards Contingency Plan provides a description of SECURITY EVENTS indicative of a potential loss of the level of safety of the plant.

NOTE: DO NOT revise this Technical Basis Document to add any identifying information to any SECURITY EVENT codes, and do not remove this note.

Threats under this EAL include the following SECURITY EVENT categories:

- SE-1, SE-2, SE-5, SE-10, SE-16, SE-18 & SE-19

Security Watch Commanders are the designated on-site 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 DCPP Security and Safeguards Contingency Plan (Safeguards) information. (ref. 1).

~~EAL #1~~The first threshold:

I~~is~~s applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA (OCA). This includes any action directed against an ISFSI that is located outside the PLANT PROTECTED AREA.

This event will require rapid response and assistance due to the possibility of the attack progressing to the PLANT PROTECTED AREA.

~~EAL #2~~The OCA is the area and boundary contained in the DCPP Security and Safeguards Contingency Plan (ref. 1). Generally described, it is the area between Security Gate A (aka North Gate, and is located on the road located at the north edge of the exclusion area/SITE BOUNDARY) to Security Gate E (located on the main access road just north of Secondary (Backup) Met Tower and the SITE BOUNDARY), and extending eastward to encompass the 500 and 230kV switchyards, and bounded on the west by the Pacific Ocean. On UFSAR Figure 2.1-2 this is approximated as the "Exclusion Area Boundary". A copy of UFSAR Figure 2.1-2 is at the end of definitions section of this document.

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 as outlined in DCPP Administrative Procedure XI1.ID2 (ref. 3).

The second threshold:

An assessment of ~~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~~)~~.

This event will require rapid response and assistance due to the possibility of the need to prepare the plant and staff for a potential aircraft impact.

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.

Background

The security shift supervision is defined as the Security Watch Commander (ref. 1).

Timely and accurate communications between the Security ~~Shift Supervision~~ Watch Commander 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.

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 DCPP Security and Safeguards Contingency Plan~~Security Plan~~.

DCPP Basis Reference(s):

1. DCPP Security and Safeguards Contingency Plan
2. DCPP Procedures (Procedure names and designations are controlled due to the nature of Safeguards and 10 CFR § 2.39 information.)
3. DCPP Administrative Procedure XI1.ID2 "Regulatory Reporting Requirements and Reporting Process"
4. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the PLANT PROTECTED AREA.

EAL:

HS1.1 **Site Area Emergency**
A HOSTILE ACTION is occurring or has occurred within the PLANT PROTECTED AREA as reported by the Security Watch Commander.

Mode Applicability:

All

Definition(s):

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

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

OWNER CONTROLLED AREA (OCA) - For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan.

PLANT PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

Basis:

<u>ERO Decision Making Information</u>

<u>The intent of this EAL is to address the potential for a very rapid progression of events due to a dedicated attack. It is not intended to address incidents that are accidental or acts of civil disobedience, such as physical disputes between employees within the OCA or PLANT PROTECTED AREA. Those events are adequately addressed by other EALs. HOSTILE ACTION identified above encompasses various acts including SECURITY EVENTS:</u>

<u>NOTE: DO NOT revise this Technical Basis Document to add any identifying information to any SECURITY EVENT codes, and do not remove this note.</u>

<u>Threats under this EAL include the following SECURITY EVENT categories:</u>

- SE-2, SE-4, SE-5, SE-10, SE-16

This class of SECURITY EVENTS represents an escalated threat to plant safety above that contained in the Alert IC in that a hostile force has progressed from the OWNER CONTROLLED AREA (OCA) to the PLANT PROTECTED AREA (PA). Although DCPP security officers are well trained and prepared to protect against hostile action, it is appropriate for Offsite Response Organizations (OROs) to be notified and encouraged to begin preparations for public protective actions (if they do not normally) to be better prepared should it be necessary to consider further actions.

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

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 as outlined in DCPP Administrative Procedure XI1.ID2 (ref. 3).

Security Watch Commanders are the designated on-site 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 DCPP Security and Safeguards Contingency Plan (Safeguards) information. (ref. 1).

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

Background

The security shift supervision is defined as the Security Watch Commander.

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

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

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

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 DCPP Security and Safeguards Contingency Plan~~Security Plan~~.

DCPP Basis Reference(s):

1. DCPP Security and Safeguards Contingency Plan
2. DCPP Procedures (Procedure names and designations are controlled due to the nature of Safeguards and 10 CFR §.2.39 information.)
3. DCPP Administrative Procedure XI1.ID2 "Regulatory Reporting Requirements and Reporting Process"
4. NEI 99-01 HS1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than Design Earthquake (DE) level.

EAL:

HU2.1	**Unusual Event**
Seismic event > DE PGA as indicated by ground acceleration > 0.2 g on the "X" or "Y" axis or > 0.133 g on the "Z" axis. (Note 11)	

Note 11: If the Earthquake Force Monitor (EFM) is out of service, refer to CP M-4 Earthquake for alternative methods to assess earthquakes.

Mode Applicability:

All

Definition(s):

None

Basis:

ERO Decision Making Information

Ground motion acceleration > 0.2 g on the "X" or "Y" axis or > 0.133g on the "Z" axis is the peak ground acceleration (PGA) criterion for a Design Earthquake (DE) (ref. 3).

If the EFM indicator alarms (> 0.2 g on the "X" or "Y" axis or > 0.133g on the "Z" axis) indicating the DE PGA has been exceeded, an Unusual Event should be declared. The "X" and "Y" axes correspond to horizontal peak acceleration values while the "Z" axis corresponds to vertical peak acceleration values.

If the EFM is not operable, the earthquake magnitude is determined by alternative methods in accordance with CP M-4, "Earthquake." If it is determined that any peak acceleration has exceeded 0.2 g on the "X" or "Y" axis or 0.133g on the "Z" axis, an Unusual Event should be declared (ref. 3).

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

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

Background

In the event of an earthquake measuring greater than or equal to 0.01 g, the Seismic Instrumentation System annunciator PK15-24 will alert the control room and peak acceleration indications will be displayed on the EFM. The primary means for timely determination of the magnitude of an earthquake, and subsequently assessing emergency action levels, is using the EFM located in the control room (ref. 2).

When the seismic monitoring system alarms, SM directs actions as defined in CP M-4, "Earthquake," and the seismic instrumentation system engineer is notified to coordinate post-earthquake activities including retrieval and analysis of the seismic event data. The purpose of the analysis is to determine within 4 hours whether the computed response spectra associated with any of the three directional components of the seismic event exceed the DE response spectra exceedance criterion (ref.4).

It should be noted that the DE PGA values are the zero period accelerations associated the DE response spectra. Since the DE PGA indications are available and displayed on the EFM within minutes, these are the indications used for timely emergency classification. The seismic monitoring system also stores the seismic event data and generates reports later used during the post-earthquake evaluation (ref.4)

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. Such confirmation should not, however, preclude a timely emergency declaration based on receipt of the EFM alert alarm. 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 DCPP. Alternatively, near real-time seismic activity can be accessed via the NEIC website:

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

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for ~~an Operating Basis~~Design Earthquake (~~OBE~~DE). An earthquake greater than ~~an OBE~~ DE but less than a ~~Safe Shutdown~~ Double Design Earthquake (~~SSE~~DDE) 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.

NOTE: An Operating Basis Earthquake (OBE) is referred to as Design Earthquake (DE) at DCPP, and a Safe Shutdown Earthquake (SSE) is referred to as Double Design Earthquake (DDE) at DCPP (ref. 3).

DCPP Basis Reference(s):

1. DCM T-6, Seismic Analysis of Structures
2. AR PK 15-24, Seismic Instr System
3. CP M-4, Earthquake
4. AWP E-017 Guidelines for Post-Earthquake Engineering Response
5. 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 PLANT PROTECTED AREA.	

Mode Applicability:

All

Definition(s):

PLANT PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

TORNADO - A violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

Basis:

ERO Decision Making Information

A TORNADO striking (touching down) within the PLANT PROTECTED AREA warrants declaration of an Unusual Event regardless of the measured wind speed at the meteorological tower.

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.

Background

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 PLANT PROTECTED AREA.

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

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

~~EAL #4 addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal~~

~~vehicles. Examples of such an event include site flooding caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.~~

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

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

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

DCPP Basis Reference(s):

1. CP M-16 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 required for the current operating mode. (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.

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:

ERO Decision Making Information

~~EAL #2~~ 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.

In modes 5, 6 and defueled, the appropriate plant configuration based Outage Safety Checklist in AD8.DC55 "Outage Safety Scheduling" should be consulted to identify required equipment supporting each of the specified safety functions (ref. 1).

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

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

Background

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.~~

~~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).~~

DCPP Basis Reference(s):

1. AD8.DC55 Outage Safety Scheduling
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:

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

Mode Applicability:

All

Definition(s):plant

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

PLANT PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

Basis:

ERO Decision Making Information

This EAL is applicable to events in areas external to the DCPP PLANT PROTECTED AREA.

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

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

Background

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 #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).~~

DCPP Basis Reference(s):

1. CP M-9A Hazardous Material Incident – Initial Emergency Response/Mitigation Procedure
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:

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

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

Mode Applicability:

All

Definition(s):

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

Basis:

ERO Decision Making Information

~~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 when both north and south access routes are unavailable due to site FLOODING caused by a hurricane, heavy rains, ~~up-river water releases,~~ dam failure, tsunami, mudslide, etc., ~~or an on-site train derailment~~ blocking the access and egress roads (refer to CP M-12).

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

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

Background

Refer to CP M-12 Stranded Plant for conditions in which viable plant access routes are lost (ref. 1).

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.~~

DCPP Basis Reference(s):

1. CP M-12 Stranded Plant
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.1 Unusual Event

A FIRE is **not** extinguished within 15 minutes 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 SM/SEC/ED should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Containment • Auxiliary Building • Fuel Handling Building • Turbine Building • Intake Structure Lower Levels • Pipe Rack • Main, Auxiliary & Startup Transformers

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.

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:

ERO Decision Making Information

Multiple flow switches for the same general vicinity constitute a single alarm. This is because water flow in the sprinkler system can be seen on multiple switches for the same location. However, smoke and flame detectors are all individual alarms spaced far enough apart that each should be considered independent of each other.

~~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 HU4.1 assessment purposes, the emergency declaration clock starts at the time that multiple alarms or indications are received, the report was received, or the time that a single alarm is confirmed by subsequent verification action.~~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~~

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

Background

Table H-1 Fire Areas are based on CP M-10, Fire Protection of Safe Shutdown Equipment. 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~~

~~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.~~

DCPP Basis Reference(s):

1. CP M-10, Fire Protection of Safe Shutdown Equipment

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 associated with **any** Table H-1 area.

AND

The existence of a FIRE is **not** verified within 30 minutes of alarm receipt. (Note 1)

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

Table H-1 Fire Areas
• Containment • Auxiliary Building • Fuel Handling Building • Turbine Building • Intake Structure Lower Levels • Pipe Rack • Main, Auxiliary & Startup Transformers

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.

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:

ERO Decision Making Information

Multiple flow switches for the same general vicinity constitute a single alarm. This is because water flow in the sprinkler system can be seen on multiple switches for the same location. However, smoke and flame detectors are all individual alarms spaced far enough apart that each should be considered independent of each other.

An "Incipient Alarm" meets the intent of a "single fire alarm." A "pre-alarm" does not meet the intent of a "single fire alarm."

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 HU4.2 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.

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.

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

Background

Table H-1 Fire Areas are based on CP M-10, Fire Protection of Safe Shutdown Equipment. 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 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~~

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.

~~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~~

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~~HU4.2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

DCPP Basis Reference(s):

1. CP M-10, Fire Protection of Safe Shutdown Equipment
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 ISFSI PROTECTED AREA or PLANT PROTECTED AREA **not** extinguished within 60 minutes of the initial report, alarm or indication. (Note 1)	

Note 1: The SM/SEC/ED 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.

ISFSI PROTECTED AREA - Areas within the ISFSI to which access is strictly controlled in accordance with the station's Security Plan.

PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

Basis:

ERO Decision Making Information

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 ~~#1~~HU4.1 or ~~EAL #2~~HU4.2, a FIRE within the PLANT PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety. This basis extends to a FIRE occurring within the Protected Area of ~~an~~ the ISFSI located outside the PLANT PROTECTED AREA. ~~[*Sentence for plants with an ISFSI 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.

Background

None~~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~~

~~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.~~

DCPP 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 ISFSI PROTECTED AREA or PLANT PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.	

Mode Applicability:

All

Definition(s):

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

ISFSI PROTECTED AREA - Areas within the ISFSI to which access is strictly controlled in accordance with the station's Security Plan.

PLANT PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

Basis:

ERO Decision Making Information

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 ~~[*for plants with an ISFSI outside the plant Protected Area*]~~ or PLANT PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department, for DCPP this is normally CalFire), 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 (engages in firefighting efforts or is needed to engage in firefighting efforts) because the fire is beyond the capability of the Fire Brigade (for DCPP, this is the DCPP Fire Department) 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.

Background

None

~~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~~

~~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.~~

DCPP 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 Rooms/Areas	
Room/Area	**Mode(s)**
Auxiliary Building – 115' - BASTs	2, 3, 4
Auxiliary Building – 100' – BA Pumps	2, 3, 4
Auxiliary Building – 85' – Aux Control Board	2, 3, 4
Auxiliary Building – 64' – BART Tank area	2, 3, 4
Area H (below Control Room) – 100' 480V Bus area/rooms	3, 4

Mode Applicability:

2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown

Definition(s):

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

Basis:

ERO Decision Making Information

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. Specifically, the identified rooms are those where an activity must be performed to borate to cold shutdown, isolate accumulators or cooldown using RHR.

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

An 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~~SM/SAC/ED 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.

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)~~. Such events are classified per IC HU4 - Fire.

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

Background

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

An 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.

NOTE*: IC HA5 mode applicability has been limited to the applicable modes identified in Table H-2 Safe Operation & Shutdown Rooms/Areas. If due to plant operating procedure or plant configuration changes, the applicable plant modes specified in Table H-2 are changed, a corresponding change to Attachment 3 'Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases' and to IC HA5 mode applicability is required.*

DCCP Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Rooms/Areas Table 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 requiring plant control to be transferred from the Control Room to the Hot Shutdown Panel area.	

Mode Applicability:

All

Definition(s):

None

Basis:

ERO Decision Making Information

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

The Shift Manager (SM) determines if the Control Room requires evacuation and entry into OP AP-8A. 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. OP AP-8A Control Room Inaccessibility – Establishing Hot Standby and OP AP-8B Control Room Inaccessibility – Hot Standby to Cold Shutdown provides the instructions establishing plant control from outside the Control Room (Ref. 1, 2).

Inability to establish plant control from outside the Control Room escalates this event to a Site Area Emergency per EAL HS6.1.

Escalation of the emergency classification level would be via IC HS6.

Background

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.

DCPP Basis Reference(s):

1. OP AP-8A Control Room Inaccessibility – Establishing Hot Standby

2. OP AP-8B Control Room Inaccessibility – Hot Standby to Cold Shutdown
3. OP AP-34.5.1 Fire Response – Cable Spreading Room (FA 7-A)
4. OP AP-34.5.3 Fire Response – Control Room (CR-1)
5. 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 Hot Shutdown Panel area.

AND

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

- Reactivity (Modes 1, 2 and 3 **only**)
- Core Cooling
- RCS heat removal

Note 1: The SM/SEC/ED 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:

ERO Decision Making Information

The Shift Manager (SM) determines if the Control Room requires evacuation per OP AP-8A. 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. OP AP-8A Control Room Inaccessibility – Establishing Hot Standby and OP AP-8B Control Room Inaccessibility – Hot Standby to Cold Shutdown, provides the instructions establishing plant control from outside the Control Room (Ref. 1, 2).

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on ~~Emergency Director~~SM/SEC/ED judgment. The ~~Emergency Director~~SM/SEC/ED 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). The 15 minute clock starts once plant control has been transferred to the Hot Shutdown Area (OP AP-8A Attachment 4 480V Bus Alignment and Appendix F Electrical System Actions).

Physical control of key safety functions by manipulation of controls is **not** required to verify control, rather, it is sufficient that control transfer is successful (i.e. light indication of applicable equipment).

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

Background

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 shut down the reactor and maintain it shutdown), RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink).

Hot Shutdown Panel (HSDP) indications for Reactivity, Core Cooling and RCS Heat Removal:

- Reactivity
  - Gamma Metrics indicators (NI-53 & NI-54)
- Core Cooling
  - Pressurizer Liquid Temperature (TI-453B)
  - Pressurizer Pressure (PI-455B)
  - RCS WR Pressure (PI-406 at Dedicated Shutdown Panel)
  - RCS Temperatures (Loop 1 at Dedicated Shutdown Panel)
- RCS heat removal
  - AFW Flow Indicators (FI-165 through 168)
  - AFW Pump discharge pressures (PI-51B through 53B)
  - SG WR Levels (LI-501 through 504)
  - SG Pressures (PI-514, 524, 534, 544)

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.

DCPP Basis Reference(s):

1. OP AP-8A Control Room Inaccessibility – Establishing Hot Standby
2. OP AP-8B Control Room Inaccessibility – Hot Standby to Cold Shutdown
3. OP AP-34.5.1 Fire Response – Cable Spreading Room (FA 7-A)
4. OP AP-34.5.3 Fire Response – Control Room (CR-1)
5. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SM/SEC/ED Judgment

Initiating Condition: Other conditions existing that in the judgment of the SM/SEC/ED warrant declaration of a UE.

EAL:

HU7.1	**Unusual Event**
Other conditions exist which in the judgment of the SM/SEC/ED 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:

ERO Decision Making Information

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

Background

The SM/SEC/ED is the designated onsite individual having the responsibility and authority for implementing the DCPP Radiological Emergency Response Plan. The operations Shift Manager (SM) initially acts in the capacity of the SEC/ED and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the SM/SEC/ED, 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.

Refer to CP M-10 Fire Protection of Safe Shutdown Equipment for a list of SAFETY SYSTEMS.

DCPP Basis Reference(s):

1. NEI 99-01 HU7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SM/SEC/ED Judgment

Initiating Condition: Other conditions exist that in the judgment of the SM/SEC/ED warrant declaration of an Alert.

EAL:

HA7.1	**Alert**

Other conditions exist which, in the judgment of the SM/SEC/ED, 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):

EPA PROTECTIVE ACTION GUIDELINES (EPA PAG) - The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs requires DCPP to recommend protective actions for the general public to offsite planning agencies.

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

OWNER CONTROLLED AREA (OCA) - For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan.

SECURITY EVENT - Any incident representing an attempted, threatened, or actual breach of the security system or reduction of the operational effectiveness of that system. A security event can result in either a SECURITY CONDITION or HOSTILE ACTION.

Basis:

ERO Decision Making Information

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

Background

The SM/SEC/ED is the designated onsite individual having the responsibility and authority for implementing the DCPP Radiological Emergency Response Plan. The operations Shift Manager (SM) initially acts in the capacity of the SEC/ED and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the SM/SEC/ED, 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.

DCPP Basis Reference(s):

1. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SM/SEC/ED Judgment

Initiating Condition: Other conditions existing that in the judgment of the SM/SEC/ED warrant declaration of a Site Area Emergency.

EAL:

HS7.1	**Site Area Emergency**
Other conditions exist which in the judgment of the SM/SEC/ED 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):

EPA PROTECTIVE ACTION GUIDELINES (EPA PAG) - The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs requires DCPP to recommend protective actions for the general public to offsite planning agencies.

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

OWNER CONTROLLED AREA (OCA) - For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan.

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

ERO Decision Making Information

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

Background

The SM/SEC/ED is the designated onsite individual having the responsibility and authority for implementing the DCPP Radiological Emergency Response Plan. The operations Shift Manager (SM) initially acts in the capacity of the SEC/ED and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the SM/SEC/ED, 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.

DCPP Basis Reference(s):

1. NEI 99-01 HS7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SM/SEC/ED Judgment

Initiating Condition: Other conditions exist which in the judgment of the SM/SEC/ED warrant declaration of a General Emergency.

EAL:

HG7.1	**General Emergency**
Other conditions exist which in the judgment of the SM/SEC/ED 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):

EPA PROTECTIVE ACTION GUIDELINES (EPA PAG) - The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs requires DCPP to recommend protective actions for the general public to offsite planning agencies.

HOSTILE ACTION - An act toward DCPP 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 DCPP. 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.

OWNER CONTROLLED AREA (OCA) - For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan.

Basis:

ERO Decision Making Information

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

Releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the SITE BOUNDARY.

Background

The SM/SEC/ED is the designated onsite individual having the responsibility and authority for implementing the DCPP Radiological Emergency Response Plan. The operations Shift Manager (SM) initially acts in the capacity of the SEC/ED and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the SM/SEC/ED, 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.

DCPP Basis Reference(s):

1. NEI 99-01 HG7

Category S – System Malfunction

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

Numerous system-related equipment failure events that warrant emergency classification have been identified in this category. They may pose actual or potential threats to plant safety.

The events of this category pertain to the following subcategories:

1. Loss of Vital AC Power

Loss of vital 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 vital buses.

2. Loss of Vital DC Power

Loss of vital 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 125 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. RTS Failure

This subcategory includes events related to failure of the Reactor Trip System (RTS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RTS 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 RTS 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.

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.

Category: S – System Malfunction

Subcategory: 1 – Loss of Vital AC Power

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

EAL:

SU1.1 Unusual Event

Loss of **all** offsite AC power capability, Table S-1, to Unit 1 or Unit 2 vital 4.16KV buses F, G and H for ≥ 15 minutes. (Note 1)

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

Table S-1 AC Power Capability		
	Unit 1	**Unit 2**
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• Aux XFMR 1-2 fed from the Main Generator • DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• Aux XFMR 2-2 fed from the Main Generator • DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

Mode Applicability:

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

Definition(s):

None

Basis:

ERO Decision Making Information

For emergency classification purposes, "capability" means that, whether or not the buses are actually powered from it, an AC power source(s) can be aligned to the vital buses within 15 minutes:

- By a clear procedure path,
  and
- Breakers and equipment are readily available to power up the bus within the allotted time frame.

The vital 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant (see figure below).

This ~~IC~~ EAL addresses a prolonged (greater than 15 minutes) 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.

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

When filling out the ENF form, this event can be Unit 1, Unit 2 or Unit 1 and 2.

Escalation of the emergency classification level would be via IC SA1.

Background

Unit 1(2) 4.16KV buses F, G and H are the emergency (vital) buses. Each bus has three sources of power.

One offsite source is from the 230KV switchyard via 12KV startup transformer 1-1 (2-1) to the 4.16KV startup transformer 1-2 (2-2).

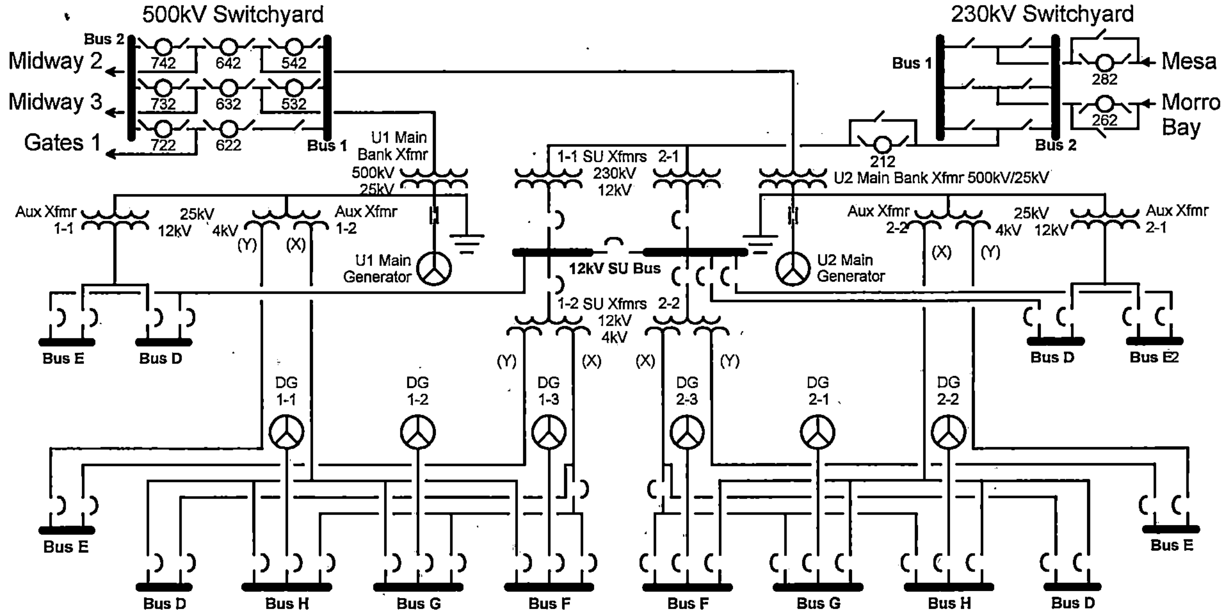
If the 1-1- or 2-1 transformer is unavailable, the other unit's #1 transformer (2-1 or 1-1) can be used to supply the startup bus through the startup bus cross tie breaker.

Another method to obtain offsite power is by back feeding the vital buses from the 500KV switchyard through the main transformer to the 4.16KV unit auxiliary transformer 1-2 (2-2). This is normally only done post-trip when 500KV power is available.

During normal operations, vital bus power is supplied from onsite by the main generator via the 4.16KV unit auxiliary transformer 1-2 (2-2). In addition, each units vital buses F, G and H have an onsite emergency diesel generator which can supply electrical power to its associated bus automatically in the event that the preferred source becomes unavailable (ref. 1-6).

~~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.~~

DCPP Electrical Distribution System



DCPP Basis Reference(s):

1. UFSAR, Section 8.2.2
2. UFSAR, Section 8.3.1.6
3. OP AP SD-1, Loss of AC Power
4. OP AP-2, Loss of Offsite Power
5. OP J-2:V, Backfeeding the Unit From the 500kV System
6. ECA-0.0, Loss of All Vital AC Power
7. NEI 99-01 SU1

Category: S – System Malfunction

Subcategory: 1 – Loss of Vital AC Power

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

EAL:

SA1.1 Alert

AC power capability, Table S-1, to Unit 1 or Unit 2 vital 4.16KV buses F, G and H reduced to a single power source for ≥ 15 minutes. (Note 1)

AND

A failure of that single power source will result in loss of **all** AC power to SAFETY SYSTEMS.

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

Table S-1 AC Power Capability

	Unit 1	Unit 2
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• Aux XFMR 1-2 fed from the Main Generator • DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• Aux XFMR 2-2 fed from the Main Generator • DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

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:

ERO Decision Making Information

For emergency classification purposes, "capability" means that, whether or not the buses are actually powered from it, an AC power source(s) can be aligned to the vital buses within 15 minutes:

- By a clear procedure path,
  and
- Breakers and equipment are readily available to power up the bus within the allotted time frame.

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

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

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

When filling out the ENF form, this event can be Unit 1, Unit 2 or Unit 1 and 2.

Escalation of the emergency classification level would be via IC SS1.

Background

The vital 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant (see figure below).Unit 1(2) 4.16KV buses F, G and H are the emergency (vital) buses. Each bus has three sources of power.

One offsite source is from the 230KV switchyard via 12KV startup transformer 1-1 (2-1) to the 4.16KV startup transformer 1-2 (2-2).

If the 1-1- or 2-1 transformer is unavailable, the other unit's #1 transformer (2-1 or 1-1) can be used to supply the startup bus through the startup bus cross tie breaker.

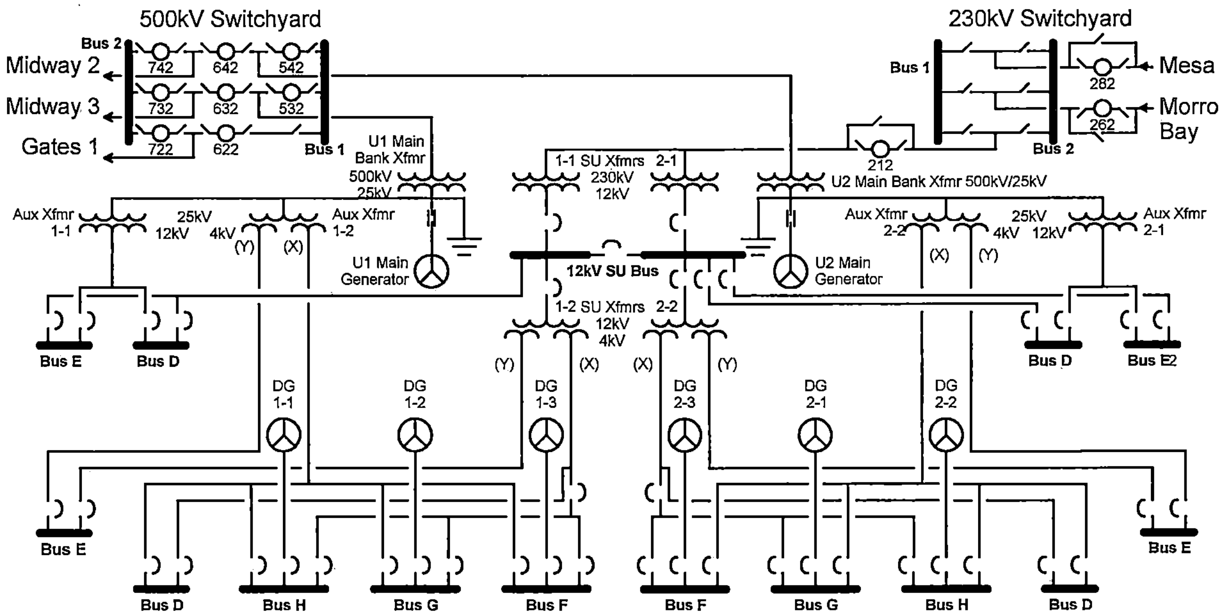
Another method to obtain offsite power is by back feeding the vital buses from the 500KV switchyard through the main transformer to the 4.16KV unit auxiliary transformer 1-2 (2-2). This is normally only done post-trip when 500KV power is available.

During normal operations, vital bus power is supplied from onsite by the main generator via the 4.16KV unit auxiliary transformer 1-2 (2-2). In addition, each units vital buses F, G and H have an onsite emergency diesel generator which can supply electrical power to its associated bus automatically in the event that the preferred source becomes unavailable (ref. 1-6).

If the capability of a second source of emergency bus power is not restored within 15 minutes, an Alert is declared under this EAL.

Refer to CP M-10 Fire Protection of Safe Shutdown Equipment for a list of SAFETY SYSTEMS.

DCPP Electrical Distribution System



DCPP Basis Reference(s):

1. UFSAR, Section 8.2.2
2. UFSAR, Section 8.3.1.6
3. OP AP SD-1, Loss of AC Power
4. OP AP-2, Loss of Offsite Power
5. OP J-2:V, Backfeeding the Unit From the 500kV System
6. ECA-0.0, Loss of All Vital AC Power
7. NEI 99-01 SA1

Category: S – System Malfunction

Subcategory: 1 – Loss of Vital AC Power

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

EAL:

SS1.1 Site Area Emergency

Loss of **all** offsite and **all** onsite AC power capability, Table S-1, to Unit 1 or Unit 2 vital 4.16KV buses F, G and H for ≥ 15 minutes. (Note 1)

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

Table S-1 AC Power Capability

	Unit 1	Unit 2
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• Aux XFMR 1-2 fed from the Main Generator • DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• Aux XFMR 2-2 fed from the Main Generator • DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

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:

ERO Decision Making Information

For emergency classification purposes, "capability" means that, whether or not the buses are actually powered from it, an AC power source(s) can be aligned to the vital buses within 15 minutes:

- By a clear procedure path,
  and
- Breakers and equipment are readily available to power up the bus within the allotted time frame.

The 15-minute interval begins when both offsite and onsite AC power capability are lost.

This ~~IC~~ EAL addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

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

When filling out the ENF form, this event can be Unit 1, Unit 2 or Unit 1 and 2.

Escalation of the emergency classification level would be via ICs ~~AG1~~RG1, FG1 or SG1.

Background

The vital 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant (see figure below).

Unit 1(2) 4.16KV buses F, G and H are the emergency (vital) buses. Each bus has three sources of power.

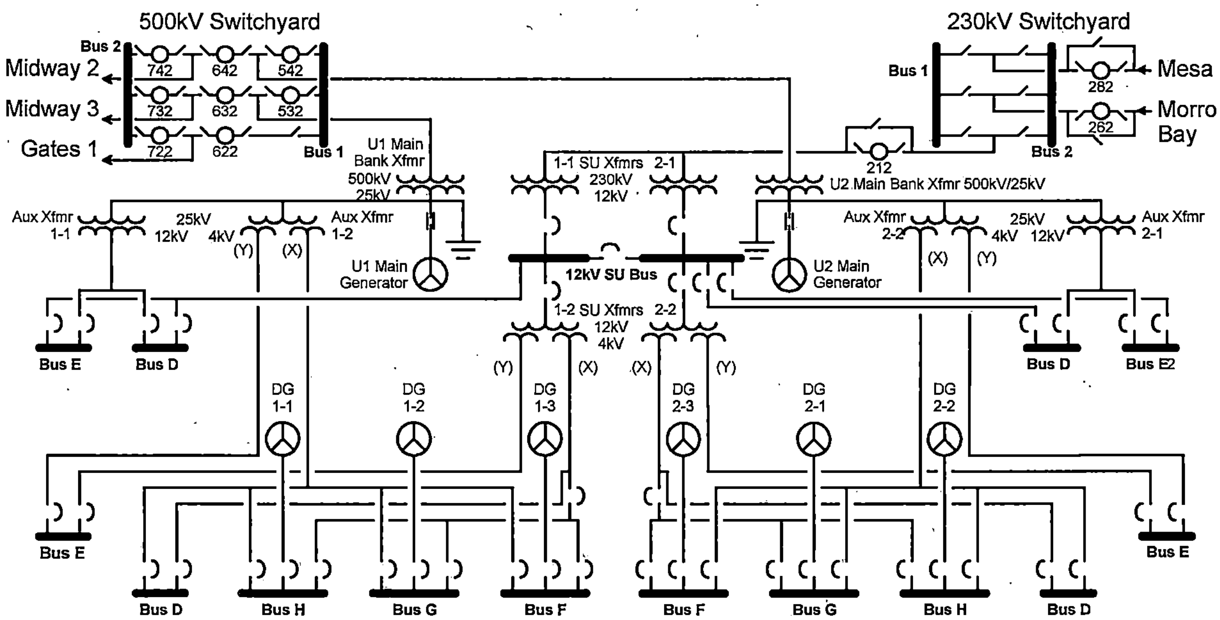
One offsite source is from the 230KV switchyard via 12KV startup transformer 1-1 (2-1) to the 4.16KV startup transformer 1-2 (2-2).

If the 1-1- or 2-1 transformer is unavailable, the other unit's #1 transformer (2-1 or 1-1) can be used to supply the startup bus through the startup bus cross tie breaker.

Another method to obtain offsite power is by back feeding the vital buses from the 500KV switchyard through the main transformer to the 4.16KV unit auxiliary transformer 1-2 (2-2). This is normally only done post-trip when 500KV power is available.

During normal operations, vital bus power is supplied from onsite by the main generator via the 4.16KV unit auxiliary transformer 1-2 (2-2). In addition, each units vital buses F, G and H have an onsite emergency diesel generator which can supply electrical power to its associated bus automatically in the event that the preferred source becomes unavailable (ref. 1-6).

DCPP Electrical Distribution System



DCPP Basis Reference(s):

1. UFSAR, Section 8.2.2
2. UFSAR, Section 8.3.1.6
3. OP AP SD-1, Loss of AC Power
4. OP AP-2, Loss of Offsite Power
5. OP J-2:V, Backfeeding the Unit From the 500kV System
6. ECA-0.0, Loss of All Vital AC Power
7. NEI 99-01 SS1

Category: S –System Malfunction

Subcategory: 1 – Loss of Vital AC Power

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

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power capability, Table S-1, to Unit 1 or Unit 2 vital 4.16KV buses F, G and H.

AND EITHER:

- Restoration of at least one 4.16KV vital bus in < 4 hours is **not** likely. (Note 1)
- CSFST Core Cooling RED path conditions met.

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

Table S-1 AC Power Capability

	Unit 1	Unit 2
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• Aux XFMR 1-2 fed from the Main Generator • DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• Aux XFMR 2-2 fed from the Main Generator • DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

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:

ERO Decision Making Information

This EAL is indicated by the extended loss of all offsite and onsite AC power capability to 4.16KV vital buses F, G and H either for greater then the DCPP Station Blackout (SBO) coping analysis time (4 hrs.) (ref. 1) 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. 2).

For emergency classification purposes, "capability" means that, whether or not the buses are actually powered from it, an AC power source(s) can be aligned to the vital buses within 15 minutes:

- By a clear procedure path,
  and
- Breakers and equipment are readily available to power up the bus within the allotted time frame.

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.

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.

When filling out the ENF form, this event can be Unit 1, Unit 2 or Unit 1 and 2.

Background

The vital 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant (see figure below).

Unit 1(2) 4.16KV buses F, G and H are the emergency (vital) buses. Each bus has three sources of power.

One offsite source is from the 230KV switchyard via 12KV startup transformer 1-1 (2-1) to the 4.16KV startup transformer 1-2 (2-2).

If the 1-1 or 2-1 transformer is unavailable, the other unit's #1 transformer (2-1 or 1-1) can be used to supply the startup bus through the startup bus cross tie breaker.

Another method to obtain offsite power is by back feeding the vital buses from the 500KV switchyard through the main transformer to the 4.16KV unit auxiliary transformer 1-2 (2-2). This is normally only done post-trip when 500KV power is available.

During normal operations, vital bus power is supplied from onsite by the main generator via the 4.16KV unit auxiliary transformer 1-2 (2-2). In addition, each units vital buses F, G and H have an onsite emergency diesel generator which can supply electrical power to its associated bus automatically in the event that the preferred source becomes unavailable (ref. 3-8).

Four hours is the station blackout coping time (ref 1).

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on SM/SEC/ED 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 (ref.2). Specifically, Core Cooling RED Path conditions exist if either:

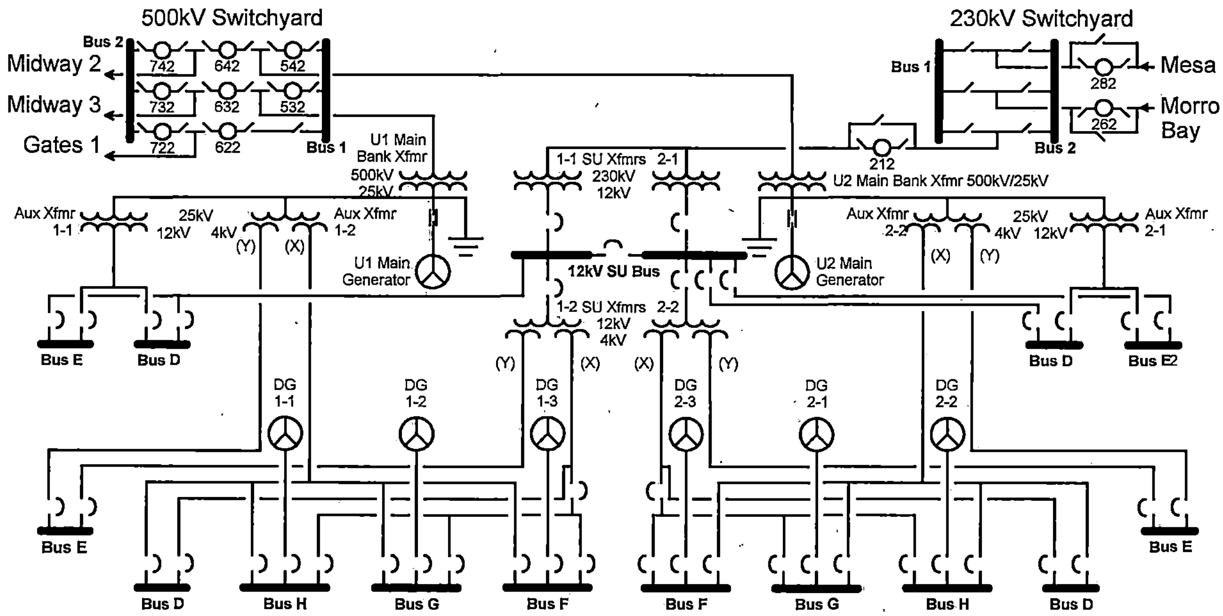
- Core exit TCs are reading greater than or equal to 1200°F, or
- Core exit TCs are reading greater than or equal to 700°F with RCS subcooling less than or equal to 20°F, and RVLIS full range indication is less than or equal 32%.

This ~~IC~~ EAL 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.

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.

DCPP Electrical Distribution System

ATTACHMENT 1
EAL Bases



DCPP Basis Reference(s):

1. DCM T-42, Station Blackout
2. F-0, Critical Safety Function Status Trees Attachment 2, Core Cooling
3. UFSAR, Section 8.2.2
4. UFSAR, Section 8.3.1.6
5. OP AP SD-1, Loss of AC Power
6. OP AP-2, Loss of Offsite Power
7. OP J-2:V, Backfeeding the Unit From the 500kV System
8. ECA-0.0, Loss of All Vital AC Power
9. 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** 125 VDC power based on battery bus voltage indications < 105 VDC on **all** Unit 1 or Unit 2 vital DC buses for ≥ 15 minutes. (Note 1)	

Note 1: The SM/SEC/ED 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):

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:

ERO Decision Making Information

Minimum battery voltage of 105 VDC is the voltage below which supplied loads may not function (ref. 1, 3, 4).

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~~SG1.

Background

The vital 125 VDC system consists of three independent networks per unit. Each network contains the following components:

- Battery
- Battery charger
- Standby battery charger to allow maintenance and/or testing

- Distribution panels
- Ground detector

The vital 125 VDC batteries and distribution panels are located in Area "H" of the 115 feet elevation of the Auxiliary Building. There are a total of three batteries per unit, 11(21), 12(22), and 13(23). The batteries are sized to provide sufficient power to operate the associated DC loads for the time necessary to safely shut down the unit, should a 480-VAC source to one or more battery chargers be unavailable (ref. 1, 2, 3).

DCPP Basis Reference(s):

1. ECA-0.0, Loss of All Vital AC Power
2. UFSAR, Section 8.3.2.2.2
3. OP AP-23, Loss of Vital DC Bus
4. Notification 50804190 DC Bus Voltage Trigger for EALs
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 capability, Table S-1, to Unit 1 or Unit 2 vital 4.16KV buses F, G and H for ≥ 15 minutes.

AND

Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **all** Unit 1 or Unit 2 vital DC buses for ≥ 15 minutes.

(Note 1)

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

Table S-1 AC Power Capability		
	Unit 1	**Unit 2**
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• Aux XFMR 1-2 fed from the Main Generator • DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• Aux XFMR 2-2 fed from the Main Generator • DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

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:

ERO Decision Making Information

For emergency classification purposes, "capability" means that, whether or not the buses are actually powered from it, an AC power source(s) can be aligned to the vital buses within 15 minutes:

- By a clear procedure path,
  and
- Breakers and equipment are readily available to power up the bus within the allotted time frame.

Minimum battery voltage of 105 VDC is the voltage below which supplied loads may not function (ref.6, 8, 9).

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

Background

This EAL is indicated by the loss of all offsite and onsite vital AC power capability to 4.16KV vital buses F, G and H 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 vital 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant (see figure below).

Unit 1(2) 4.16KV buses F, G and H are the emergency (vital) buses. Each bus has three sources of power.

One offsite source is from the 230KV switchyard via 12KV startup transformer 1-1 (2-1) to the 4.16KV startup transformer 1-2 (2-2).

Another method to obtain offsite power is by back feeding the vital buses from the 500KV switchyard through the main transformer to the 4.16KV unit auxiliary transformer 1-2 (2-2). This is normally only done post-trip when 500KV power is available.

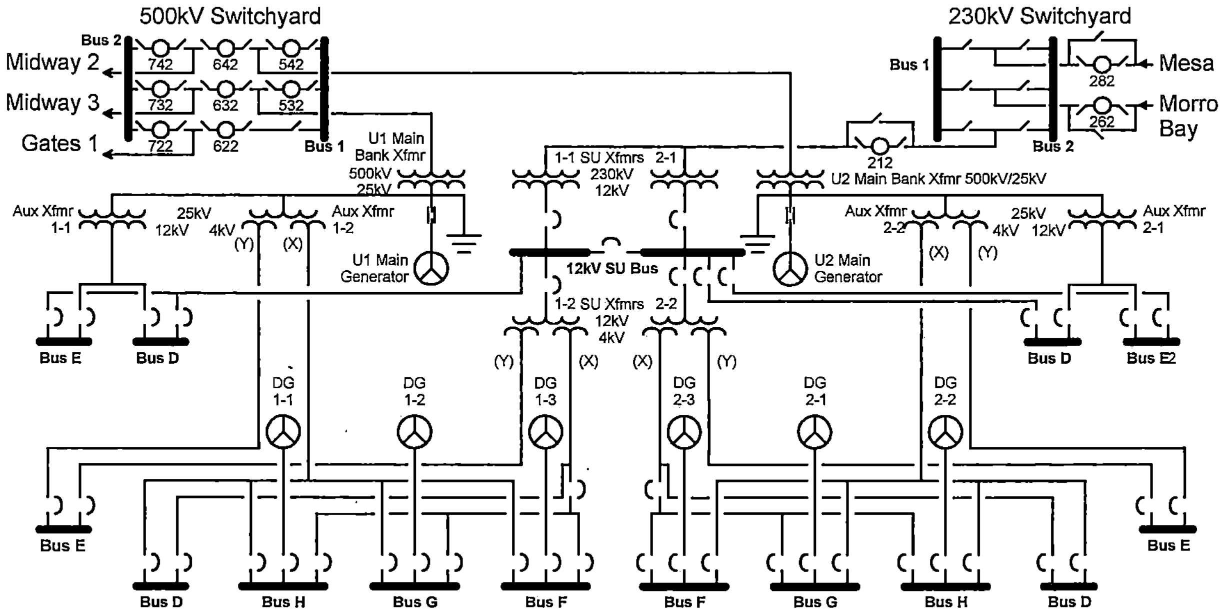
During normal operations, vital bus power is supplied from onsite by the main generator via the 4.16KV unit auxiliary transformer 1-2 (2-2). In addition, each units vital buses F, G and H have an onsite emergency diesel generator which can supply electrical power to its associated bus automatically in the event that the preferred source becomes unavailable (ref. 1-6).

The vital 125 VDC system consists of three independent networks per unit. Each network contains the following components:

- Battery
- Battery charger
- Standby battery charger to allow maintenance and/or testing
- Distribution panels
- Ground detector

The vital 125 VDC batteries and distribution panels are located in Area "H" of the 115 feet elevation of the Auxiliary Building. A total of three batteries per unit, 11(21), 12(22), and 13(23) are supplied for Units 1 and 2. The batteries are sized to provide sufficient power to operate the associated DC loads for the time necessary to safely shut down the unit, should a 480-VAC source to one or more battery chargers be unavailable (ref. 7, 8).

DCPP Electrical Distribution System



DCPP Basis Reference(s):

1. UFSAR, Section 8.2.2
2. UFSAR, Section 8.3.1.6
3. OP AP SD-1, Loss of AC Power
4. OP AP-2, Loss of Offsite Power
5. OP J-2:V, Backfeeding the Unit From the 500kV System
6. ECA-0.0, Loss of All Vital AC Power

7. UFSAR, Section 8.3.2.2.2
8. OP AP-23, Loss of Vital DC Bus
9. Notification 50804190 DC Bus Voltage Trigger for EALs
10. 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 minutes. (Note 1)

Note 1: The SM/SEC/ED 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 or emergency feed flow in at least one SG

Mode Applicability:

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

Definition(s):

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

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

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

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

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

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

Basis:

ERO Decision Making Information

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 (PPC), ERFDS and SPDS serve as a redundant compensatory indicators which may be utilized in lieu of normal Control Room indicators (ref. 1).

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.

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.

Background

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.

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.

DCPP Basis Reference(s):

1. UFSAR Section 7.5 Safety-Related Display 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 minutes. (Note 1)

AND

Any significant transient is in progress, Table S-3.

Note 1: The SM/SEC/ED 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 or emergency feed flow in at least one SG

Table S-3 Significant Transients
• Reactor trip • Runback ≥ 25% thermal power • Electrical load rejection > 25% electrical load • ECCS actuation

Mode Applicability:

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

Definition(s):

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

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

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

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

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

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

Basis:

ERO Decision Making Information

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 (PPC), ERFDS and SPDS serve 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.

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.

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

Background

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.

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.

DCPP Basis Reference(s):

1. UFSAR Section 7.5 Safety-Related Display Instrumentation
2. NEI 99-01 SA2

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification permissible limits.

EAL:

SU4.1	**Unusual Event**
RCS activity > Technical Specification Section 3.4.16 permissible limits.	

Mode Applicability:

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

Definition(s):

None

Basis:

ERO Decision Making Information

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications.

This EAL would be met if TS 3.4.16 Required Action C.1 (place plant in Mode 3 in 6 hours) or C.2 (place plant in Mode 5 in 36 hours) were not met.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category ~~A~~ R ICs.

Background

The specific iodine activity is limited to 1.0 μCi/gm Dose Equivalent I-131. However, operation with iodine specific activity levels greater than the limit is permissible, if the activity levels do not exceed 60.0 μCi/gm Dose Equivalent I-131, for more than 48 hours.

The specific Xe-133 activity is limited to ≤ 600 μCi/gm Dose Equivalent XE-133 (ref 1).

With the Dose Equivalent I-131 greater than the LCO limit of 1 μCi/gm, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is < 60.0 μCi/gm. Dose Equivalent I-131 must be restored to within limits within 48 hours. This is acceptable since it is expected that, if there were an iodine spike, the normal RCS iodine concentration (≤ 1 μCi/gm) would be restored within this time period (ref 2).

This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

DCPP Basis Reference(s):

1. DCPP Technical Specifications section 3.4.16 RCS Specific Activity
2. DCPP Technical Specifications Basis section 3.4.16 RCS Specific Activity
3. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits.

EAL:

SU4.2	**Unusual Event**
With letdown in service, procedurally directed letdown dose point radiation > 3 R/hr.	

Mode Applicability:

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

Definition(s):

None

Basis:

ERO Decision Making Information

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.

Background

Initial indication of Fuel Clad degradation can be determined by measuring the external radiation dose rate at a distance of one foot from the center of the letdown line in the letdown heat exchanger room using the technique described in Attachment 7.1 of EP RB-14A, Initial Detection of Core Damage. An external radiation dose rate exceeding 3 R/hr indicates Fuel Clad degradation greater than Technical Specification allowable limits. This value was determined by ratioing 15 R/hr which corresponds to coolant activity at 300 μCi/gm to the Technical Specification LCO coolant activity of 60 μCi/gm which includes iodine spike (see EAL SU4.1), or 15 R/hr x 60/300 = 3 R/hr (ref 1, 2, 3).

DCPP Basis Reference(s):

1. EP RB-14A, Initial Detection of Core Damage
2. DCPP Technical Specifications section 3.4.16 RCS Specific Activity
3. PG&E Calculation EP 95-02 Rev. 0, Letdown Heat Exchanger Rom Dose Rates Corresponding to EP G-1, Alert No. 2 RCS Activity
4. 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 minutes.
OR
RCS identified leakage > 25 gpm for ≥ 15 minutes.
OR
Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 minutes. (Note 1)

Note 1: The SM/SEC/ED 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):

RCS LEAKAGE – RCS leakage shall be:

a. Identified Leakage
    1. Leakage, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leak off), that is captured and conducted to collection systems or a sump or collecting tank;
    2. 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;
    3. Reactor Coolant System (RCS) leakage through a steam generator to the secondary system (primary to secondary leakage).

b. Unidentified Leakage

    All leakage (except RCP seal water injection or leak off) that is not identified leakage.

c. Pressure Boundary Leakage

    Leakage (except primary to secondary leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

f. 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).

Basis:

ERO Decision Making Information

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.

~~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) (ref. 2).

~~EAL #3~~The third condition addresses an RCS mass loss caused by an UNISOLABLE leak through an interfacing system (ref. 3, 4, 5).

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

Below is a summary of classification guidance for steam generator tube leaks:

	Affected SG is FAULTED Outside of Containment?	
Primary-to-Secondary 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

Background

Manual or computer-based methods of performing an RCS inventory balance are normally used to determine RCS LEAKAGE.

STP R-10C, Reactor Coolant System Water Inventory Balance, is performed to determine the source and flow rate of the leakage. (ref. 1).

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 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.

DCPP Basis Reference(s):

1. STP R-10C, Reactor Coolant System Water Inventory Balance
2. DCPP Technical Specifications Definitions section 1.1
3. UFSAR Section 5.2.7 Reactor Coolant Pressure Boundary Leakage Detection System
4. UFSAR Section 5.2.9 Leakage Prediction From Primary Coolant Sources Outside Containment
5. OP AP-1, Excessive Reactor Coolant System Leakage
6. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 6 – RTS 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** RTS setpoint is exceeded.

AND

A subsequent automatic trip or manual trip action taken at the control room panels (CC1, VB2 or VB5) 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:

ERO Decision Making Information

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the control room panels (CC1, VB2 or VB5):

- Reactor trip switches (CC1 and VB2)
- Deenergization of 480V Buses 13D and 13E (23D and 23E) at the Control Room vertical board (VB5)

Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as locally opening the reactor trip beakers, local deenergization of 480V Buses 13D and 13E, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 4).

Following any automatic RTS trip signal, E-0 (ref. 2) and FR-S.1 (ref. 4) prescribe insertion of redundant manual trip signals to back up the automatic RTS 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 RTS 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.

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".

Should a reactor ~~(~~trip ~~[PWR] / scram [BWR])~~signal be generated as a result of plant work (e.g., ~~RPS~~ RTS 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~~ RTS fails to automatically shut down the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the ~~(~~trip ~~[PWR] / scram [BWR])~~failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

Background

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Trip System (RTS) trip function. A reactor trip is automatically initiated by the RTS when certain continuously monitored parameters exceed predetermined setpoints (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 RTS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2, 3, 4).

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~~ RTS 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 shut down 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 shut down 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. ~~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.

DCPP Basis Reference(s):

1. DCPP Technical Specifications Section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. E-0 Reactor Trip or Safety Injection
3. F-0 Critical Safety Function Status Trees - Subcriticality
4. FR-S.1 Response to Nuclear Power Generation/ATWS
5. UFSAR Section 7.6.2.3 ATWS Mitigation Actuation Circuitry (AMSAC)
6 NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RTS 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 control room panels (CC1, VB2 or VB5) 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):

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:

ERO Decision Making Information

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the control room panels (CC1, VB2 or VB5):

- Reactor trip switches (CC1 and VB2)
- Deenergization of 480V Buses 13D and 13E (23D and 23E) at the Control Room vertical board (VB5)

Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as locally opening the reactor trip beakers, local deenergization of 480V Buses 13D and 13E (23D and 23E), 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).

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.

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".

Should a reactor ~~(~~trip ~~[PWR] / scram [BWR])~~signal be generated as a result of plant work (e.g., ~~RPS~~ RTS 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~~ RTS fails to automatically shut down the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the ~~(~~trip ~~[PWR] / scram [BWR~~failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

Background

This EAL addresses a failure of a manually initiated trip in the absence of having exceeded an automatic RTS 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 a manual 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 RTS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2, 3, 4).

This IC addresses a failure of the ~~RPS~~ RTS 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 shut down 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 shut down 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~~trip~~[*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.

~~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.

DCPP Basis Reference(s):

1. DCPP Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. E-0 Reactor Trip or Safety Injection
3. F-0 Critical Safety Function Status Trees - Subcriticality
4. FR-S.1 Response to Nuclear Power Generation/ATWS
5. UFSAR Section 7.6.2.3 ATWS Mitigation Actuation Circuitry (AMSAC)
6. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 2 – RTS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are **not** successful in shutting down the reactor.

EAL:

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 control room panels (CC1, VB2 or VB5) 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):

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:

ERO Decision Making Information

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the control room panels (CC1, VB2 or VB5):

- Reactor trip switches (CC1 and VB2)
- Deenergization of 480V Buses 13D and 13E (23D and 23E) at the Control Room vertical board (VB5)

Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as locally opening the reactor trip beakers, local deenergization of 480V Buses 13D and 13E (23D and 23E), 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).

Escalation of this event to a Site Area Emergency would be under EAL SS6.1 or SM/SEC/ED judgment.

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~~ 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.

Background

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).

On the power range scale 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).

This IC addresses a failure of the ~~RPS~~RTS 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 shut down the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the ~~RPS~~RTS.

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 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. 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.

DCPP Basis Reference(s):

1. DCPP Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. E-0 Reactor Trip or Safety Injection
3. F-0 Critical Safety Function Status Trees - Subcriticality
4. FR-S.1 Response to Nuclear Power Generation/ATWS
5. UFSAR Section 7.6.2.3 ATWS Mitigation Actuation Circuitry (AMSAC)
6. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 2 – RTS 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 conditions met. • CSFST Heat Sink RED path conditions met. **AND** Bleed and feed criteria met.

Mode Applicability:

1 - Power Operation

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:

ERO Decision Making Information

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.

Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as local deenergization of 480V Buses 13D and 13E (23D and 23E), 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, 4).

Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED path conditions being met (ref. 2).

Indication of inability to adequately remove heat from the RCS is manifested by CSFST Heat Sink RED path conditions being met in combination with bleed and feed criteria being met (ref. 3).

Heat Sink RED path conditions exist if:

- All SG narrow range levels less are than 15%, AND
- Less than 435 total AFW available to feed the SGs

Bleed and feed criteria are met when:

- Wide range level in any three (3) SGs is less than 18% [26%] AND
- All narrow range SG levels are less than 15% [25%].

Parenthetical values are used during Adverse Containment Conditions.

Escalation of the emergency classification level would be via IC ~~AG1~~ RG1 or FG1.

Background

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.

On the power range scale 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, 4).

This IC addresses a failure of the ~~RPS~~RTS 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 shut down 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 shut down the reactor.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees – Attachment 1, Subcriticality
2. F-0 Critical Safety Function Status Tress – Attachment 2, Core Cooling
3. F-0 Critical Safety Function Status Tress – Attachment 3, Heat Sink
4. FR-S.1 Response to Nuclear Power Generation/ATWS
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 offsite communication methods.

OR

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

Table S-4 Communication Methods

System	Onsite	Offsite	NRC
Unit 1, Unit 2 and TSC Radio Consoles	X	X	
DCPP Telephone System (PBX)	X	X	X
Portable radio equipment (handie-talkies)	X		
Operations Radio System	X	X	
Security Radio Systems	X		
CAS and SAS Consoles	X	X	X
Fire Radio System	X		
Hot Shutdown Panel Radio Consoles	X	X	X
Public Address System	X		
NRC FTS			X
Mobile radios	X		
Satellite phones	X	X	X
Direct line (ATL) to the County and State OES		X	

Mode Applicability:

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

Definition(s):

None

Basis:

ERO Decision Making Information

Onsite, offsite and NRC communications include one or more of the systems listed in Table S-4 (ref. 1, 2, 3).

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

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 offsite response organizations (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.).

Background

~~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 county EOCs.

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

DCPP Basis Reference(s):

1. UFSAR, Section 9.5.2
2. Emergency Plan Section 7.2 Communications Equipment
3. AR PK15-23, Communications
4. 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 minutes of a VALID containment isolation signal. (Note 1)
OR
Containment pressure ≥ 22 psig with < one full train of containment depressurization equipment operating per design for ≥ 15 minutes. (Notes 1, 9)

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

Note 9: One Containment Spray pump and two CFCUs 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:

ERO Decision Making Information

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 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. In order for a penetration to be considered isolated, a minimum of one valve in the flow path must be closed. 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. 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 or restore equipment that may not have automatically started or actuated as required, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment sprays ~~or ice condenser fans~~) are either lost or performing in a degraded manner.

This event would escalate to a Site Area Emergency in accordance with IC FS1 if there were a concurrent loss or potential loss of either the Fuel Clad or RCS fission product barriers.

Background

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases requirement. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. The Refueling Water Storage Tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, Containment Spray, if needed, is transferred to the RHR Pumps and the Containment Spray Pumps are shut down(ref. 5).

The Containment Cooling System consists of two trains of Containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement. Each train, consisting of two Containment Fan Cooling Units (CFCU), is supplied with cooling water from a separate loop of Component Cooling Water (CCW). In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically if not already running (ref.5).

The Containment pressure setpoint (22 psig, ref. 1, 2, 3, 4) is the pressure at which the equipment should actuate and begin performing its function. The design basis accident analyses and evaluations assume the loss of one Containment Spray System train and one Containment Cooling System train (ref. 5). Consistent with the design requirement, "one full train of depressurization equipment" is therefore defined to be the availability of one train of each system. If less than this equipment is operating and Containment pressure is above the actuation setpoint, the threshold is met if the required equipment cannot be started within 15 minutes.

DCPP Basis Reference(s):

1. AR PK01-18, CONTMT SPRAY ACTUATION red
2. F-0 Critical Safety Function Status Trees - Attachment 6, Containment
3. FR-Z.1 Response to High Containment Pressure
4. Technical Specifications Table 3.3.2-1
5. Technical Specifications B3.6.6 Containment Spray and Cooling Systems
6. 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 • Tsunami • Other events with similar hazard characteristics as determined by the SM/SEC/ED

Mode Applicability:

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

Definition(s):

DEGRADED PERFORMANCE – As applied to hazardous event thresholds, event damage significant enough to cause concern regarding the operability or reliability of the affected safety system train.

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.

TORNADO - A violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

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:

ERO Decision Making Information

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.

With respect to event damage caused by an equipment failure resulting in a FIRE or EXPLOSION, no emergency classification is required in response to a FIRE or EXPLOSION resulting from an equipment failure if the only safety system equipment affected by the event is that upon which the failure occurred. An emergency classification is required if a FIRE or EXPLOSION caused by an equipment failure damages safety system equipment that was otherwise functional or operable (i.e., equipment that was not the source/location of the failure). For example, if a FIRE or EXPLOSION resulting from the failure of a piece of safety system equipment causes damage to the other train of the affected safety system or another safety system, then an emergency declaration is required in accordance with this IC and EAL.

Escalation of the emergency classification level would be via IC FS1 or ~~AS1~~RS1.

Background

This condition represents an actual or potential substantial degradation of the level of safety of the plant. Due to this actual or potential substantial degradation, this condition can significantly reduce~~s~~ 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.

DCPP 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 (CMT): 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 containment 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 DCPP 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 SM/SEC/ED 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

DCPP 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 SM/SEC/ED would have greater assurance that escalation to a General Emergency is less IMMINENT.

DCPP 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

DCPP 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. CMT Radiation / RCS Activity

D. CMT Integrity or Bypass

E. SM/SEC/ED 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 "CMT 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 (CMT) 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 conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B **Inadequate Heat Removal**	1. CSFST Core Cooling-RED path conditions met	1. CSFST Core Cooling-MAGENTA path conditions met 2. CSFST Heat Sink-RED path conditions met **AND** Bleed and feed criteria met	None	1. CSFST Heat Sink-RED path conditions met **AND** Bleed and feed criteria met	None	1. CSFST Core Cooling-RED path conditions met **AND** Restoration procedures **not** effective within 15 minutes (Note 1)
C **CMT Radiation / RCS Activity**	1. Containment radiation (RM-30 or RM-31) > 300 R/hr 2. Dose equivalent I-131 coolant activity > 300 μCi/gm	None	1. Containment radiation (RM-30 or RM-31) > 40 R/hr	None	None	1. Containment radiation (RM-30 or RM-31) > 5,000 R/hr
D **CMT Integrity or Bypass**	None	None	None	None	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on SM/SEC/ED determination • UNISOLABLE pathway from Containment to the environment exists 2. Indications of RCS LEAKAGE outside of Containment	1. CSFST Containment-RED path conditions met (≥ 47 psig) 2. Containment hydrogen concentration ≥ 4% 3. Containment pressure ≥ 22 psig with < one full train of depressurization equipment operating per design for ≥ 15 minutes (Note 1, 9)
E **SM/SEC /ED Judgment**	1. **Any** condition in the opinion of the SM/SEC/ED that indicates loss of the fuel clad barrier	1. **Any** condition in the opinion of the SM/SEC/ED that indicates potential loss of the fuel clad barrier	1. **Any** condition in the opinion of the SM/SEC/ED that indicates loss of the RCS barrier	1. **Any** condition in the opinion of the SM/SEC/ED that indicates potential loss of the RCS barrier	1. **Any** condition in the opinion of the SM/SEC/ED that indicates loss of the containment barrier	1. **Any** condition in the opinion of the SM/SEC/ED 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 conditions met.

Definition(s):

None

Basis:

ERO Decision Making Information

Core Cooling RED path conditions exist if either (ref. 1, 2):

- Core exit TCs are reading greater than or equal to 1200°F, or
- Core exit TCs are reading greater than or equal to 700°F with RCS subcooling less than or equal to 20°F and RVLIS full range indication is less than or equal 32% with no RCPs running

Background

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 dedicated SPDS display system. Some of the data is also available on the PPC, but the PPC is for information only (ref. 1, 2).

This reading indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees - Attachment 2, Core Cooling
2. FR-C.1 Response to Inadequate Core Cooling
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Core Cooling-MAGENTA path conditions met.

Definition(s):

None

Basis:

ERO Decision Making Information

Core Cooling MAGENTA path conditions exist if core exit subcooling margin is less than 20°F and any of the following (ref. 2, 3):

- RVLIS full range less than or equal to 32% with no RCPs running and less than 700°F, or
- Core exit TCs reading greater than or equal to 700°F with no RCPs running with greater than 32% RVLIS full range, or
- RVLIS dynamic range level less than or equal to the specified dynamic head value with one or more RCPs running, Table F-2

Table F-2 RVLIS Values

RVLIS	No. RCPs	Level
Full Range	None	32%
Dynamic Head	4	44%
	3	30%
	2	20%
	1	14%

Background

Critical Safety Function Status Tree (CSFST) Core Cooling-MAGENTA path indicates subcooling has been lost and that some fuel clad damage may potentially occur. The CSFSTs are normally monitored using the dedicated SPDS display system. Some of the data is also available on the PPC, but the PPC is for information only (ref. 1).

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees - Attachment 2, Core Cooling
2. FR-C.1 Response to Inadequate Core Cooling
3. FR-C.2 Response to Degraded Core Cooling
4. 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 conditions met.

   AND

   Bleed and feed criteria met.

Definition(s):

None

Basis:

ERO Decision Making Information

Heat Sink RED path conditions exist if:

- All SG narrow range levels less are than 15%, AND
- Less than 435 gpm total AFW available to feed the SGs

Bleed and feed criteria are met when:

- Wide range level in any three (3) SGs is less than 18% [26%] AND
- All narrow range SG levels are less than 15% [25%].

Parenthetical values are used during Adverse Containment Conditions.

In combination with RCS Potential Loss B.1, meeting this threshold results in a Site Area Emergency.

Background

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.

The CSFSTs are normally monitored using the dedicated SPDS display system. Some of the data is also available on the PPC, but the PPC is for information only (ref. 1).

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees - Attachment 3, Heat Sink
2. 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. CMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment radiation (RM-30 or RM-31) > 300 R/hr.

Definition(s):

None

Basis:

ERO Decision Making Information

Containment radiation monitor readings greater than 300 R/hr (ref. 1) indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the Containment. This value is higher than that specified for RCS barrier Loss C.1.

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.

Background

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors RM-30 and RM-31. These monitors provide indication in the Control Room on PAM 2 with a range of 1R/hr to 1E7 R/hr (ref. 1).

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 approximately ~~range of 2% to 5%~~1.8% 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.

DCPP Basis Reference(s):

1. EP-CALC-DCPP-1602 Containment Radiation EAL Threshold Values
2. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.A

Barrier: Fuel Clad

Category: C. CMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. Dose equivalent I-131 coolant activity > 300 μCi/cc.

Definition(s):

None

Basis:

ERO Decision Making Information

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

This condition can be identified by either:

- RCS sample analysis
- EP RB-14A indications > 15 R/hr (ref. 1, 2)

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

Background

None

DCPP Basis Reference(s):

1. EP RB-14A Initial Detection of Fuel Cladding Damage
2. SPG-11 Obtaining the EP RB-14A Dose Rate
3. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.B

Barrier: Fuel Clad

Category: C. CMT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. SM/SEC/ED Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the SM/SEC/ED that indicates loss of the Fuel Clad barrier.

Definition(s):

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

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:

ERO Decision Making Information

The SM/SEC/ED 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 SM/SEC/ED 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 SM/SEC/ED~~Emergency Director~~ in determining whether the Fuel Clad barrier is lost

Background

None

DCPP Basis Reference(s):

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

Barrier: Fuel Clad

Category: E. SM/SEC/ED Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the SM/SEC/ED that indicates potential loss of the Fuel Clad barrier.

Definition(s):

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

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

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

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

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

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

Basis:

<u>ERO Decision Making Information</u>

<u>The SM/SEC/ED 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.</u>

- <u>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.</u>
- <u>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.</u>
- <u>Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The SM/SEC/ED should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.</u>

This threshold addresses any other factors that are to be used by the <u>SM/SEC/ED</u>~~Emergency Director~~ in determining whether the Fuel Clad barrier is potentially lost. The

SM/SEC/ED~~Emergency Director~~ should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Background

None

DCPP 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):

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.

RCS LEAKAGE – RCS leakage shall be:

a. Identified Leakage
   1. Leakage, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leak off), that is captured and conducted to collection systems or a sump or collecting tank;
   2. 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;
   3. Reactor Coolant System (RCS) leakage through a steam generator to the secondary system (primary to secondary leakage).

b. Unidentified Leakage

   All leakage (except RCP seal water injection or leak off) that is not identified leakage.

c. Pressure Boundary Leakage

   Leakage (except primary to secondary leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

g. 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).

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

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

NOTE: If an isolation valve could not be accessed due to local conditions (i.e. high radiation, temperature, etc.) then that would also make a leak unisolable, even though the inaccessible valve could isolate the leak.

Basis:

ERO Decision Making Information

A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED.

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.

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 will also be met.

Background

None

DCPP Basis Reference(s):

1. 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):

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.

RCS LEAKAGE – RCS leakage shall be:

a. Identified Leakage
   1. Leakage, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leak off), that is captured and conducted to collection systems or a sump or collecting tank;
   2. 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;
   3. Reactor Coolant System (RCS) leakage through a steam generator to the secondary system (primary to secondary leakage).

b. Unidentified Leakage

   All leakage (except RCP seal water injection or leak off) that is not identified leakage.

c. Pressure Boundary Leakage

   Leakage (except primary to secondary leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

h. 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).

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

NOTE: If an isolation valve could not be accessed due to local conditions (i.e. high radiation, temperature, etc.) then that would also make a leak unisolable, even though the inaccessible valve could isolate the leak.

Basis:

ERO Decision Making Information

The need to start an additional charging pump due to RCS LEAKAGE is an indication that the leak is in excess of charging pump capacity. This threshold is **not** met when an additional charging pump is started as prudent action. Rather, the threshold is met when an additional charging pump is started per conditions outlined in procedures OP AP-1 or OP AP-3, wherein RCS LEAKAGE exceeds capacity of a single charging pump with letdown isolated (ref. 1, 2).

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.

Background

None

DCPP Basis Reference(s):

1. OP AP-1 Excessive Reactor Coolant System Leakage
2. OP AP-3 Steam Generator Tube Failure
3. 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 conditions met.

Definition(s):

None

Basis:

ERO Decision Making Information

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 (PTS). —PTS results from a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized).

Background

None

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees - Attachment 4, Integrity and 4a Limit A Curve
2. FR-P.1 Response to Imminent Pressurized Thermal Shock Condition
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.B

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Heat Sink-RED path conditions met.

   AND

   Bleed and feed criteria met.

Definition(s):

None

Basis:

ERO Decision Making Information

Heat Sink RED path conditions exist if:

- All SG narrow range levels less are than 15%, AND
- Less than 435 gpm total AFW available to feed the SGs

Bleed and feed criteria are met when:

- Wide range level in any three (3) SGs is less than 18% [26%] AND
- All narrow range SG levels are less than 15% [25%].

Parenthetical values are used during Adverse Containment Conditions.

In combination with RCS Potential Loss B.1, meeting this threshold results in a Site Area Emergency.

Background

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.

The CSFSTs are normally monitored using the dedicated SPDS display system. Some of the data is also available on the PPC, but the PPC is for information only (ref. 1).

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees - Attachment 3, Heat Sink
2. FR-H.1 Response to Loss of Secondary Heat Sink
3. NEI 99-01 Inadequate Heat Removal RCS Loss 2.B

Barrier: Reactor Coolant System

Category: C. CMT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment radiation (RM-30 or RM-31) > 40 R/hr.

Definition(s):

N/A

Basis:

ERO Decision Making Information

Containment radiation monitor readings greater than 40 R/hr (ref. 1) indicate the release of reactor coolant to the Containment.

~~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.

Background

The readings assume the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal coolant activity, with iodine spiking, discharged into containment (ref. 1).

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors RM-30 and RM-31. These monitors provide indication in the Control Room on PAM 2 with a range of 1R/hr to 1E7 R/hr (ref. 1).

DCPP Basis Reference(s):

1. EP-CALC-DCPP-1602 Containment Radiation EAL Threshold Values
2. NEI 99-01 CMT Radiation / RCS Activity RCS Loss 3.A

Barrier: Reactor Coolant System

Category: B. CMT Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. SM/SEC/ED Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the SM/SEC/ED that indicates loss of the RCS barrier.

Definition(s):

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

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:

ERO Decision Making Information

The SM/SEC/ED 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 SM/SEC/ED 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 SM/SEC/ED~~Emergency Director~~ in determining whether the RCS Barrier is lost.

Background

None

DCPP Basis Reference(s):

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

Barrier: Reactor Coolant System

Category: E. SM/SEC/ED Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the SM/SEC/ED that indicates potential loss of the RCS barrier.

Definition(s):

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

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:

ERO Decision Making Information

The SM/SEC/ED 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 SM/SEC/ED 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 SM/SEC/ED~~Emergency Director~~ in determining whether the RCS Barrier is potentially lost. The

SM/SEC/ED~~Emergency Director~~ should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Background

None

DCPP 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:

ERO Decision Making Information

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.

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow ~~without an adverse effect on plant cooldown~~ meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, gland seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category ~~A~~R ICs.

The ~~emergency classification 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?	
Primary-to-Secondary 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

Background

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).

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.

There is no Potential Loss threshold associated with RCS or SG Tube Leakage.

DCPP 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 conditions met. **AND** Restoration procedures **not** effective within 15 minutes. (Note 1)

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

Definition(s):

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

Basis:

ERO Decision Making Information

The 15 minute clock starts upon entry into FR-C.1 Response to Inadequate Core Cooling (ref.2).

The restoration procedure is considered "effective" if core exit thermocouple readings are decreasing and/or if reactor vessel level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The SM/SEC/ED~~Emergency Director~~ should escalate the emergency classification level as soon as it is determined that the procedure(s) will not be effective.

Background

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 dedicated SPDS display system (ref. 1). Some of the data is also available on the PPC, but the PPC is for information only

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).

A direct correlation to status trees can be made if the effectiveness of the restoration procedures is also evaluated. If core exit thermocouple (CET) readings are greater than 1,200°F, the Fuel Clad barrier is also lost (see Fuel Clad Loss B.1).

This condition represents an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. For this condition to occur, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation in a significant fraction of core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees - Attachment 2, Core Cooling
2. FR-C.1 Response to Inadequate Core Cooling
3. NEI 99-01 Inadequate Heat Removal Containment Potential Loss 2.A

Barrier: Containment

Category: C. CMT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: C. CMT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

1. Containment radiation (RM-30 or RM-31) > 5,000 R/hr.

Definition(s):

None

Basis:

ERO Decision Making Information

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.

Background

Containment radiation monitor readings greater than 5,000 R/hr (ref. 1) indicate significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier (ref. 1).

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors RM-30 and RM-31. These monitors provide indication in the Control Room on PAM 2 with a range of 1R/hr to 1E7 R/hr (ref. 1).

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~~ associated 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.

DCPP Basis Reference(s):

1. EP-CALC-DCPP-1602 Containment Radiation EAL Threshold Values
2. NEI 99-01 CMT Radiation / RCS Activity Containment Potential Loss 3.A

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. Containment isolation is required.

   AND EITHER:

   - Containment integrity has been lost based on SM/SEC/ED determination.
   - UNISOLABLE pathway from containment to the environment exists.

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.

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

NOTE: If an isolation valve could not be accessed due to local conditions (i.e. high radiation, temperature, etc.) then that would also make a leak unisolable, even though the inaccessible valve could isolate the leak.

Basis:

ERO Decision Making Information

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold ~~1.~~A.1.

~~1.A.1~~First Bullet– 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).

Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the SM/SEC/ED~~Emergency Director~~ 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.).

~~1.A.2~~Second Bullet – 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.

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.

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.

Background

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~~.

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.

Refer to the middle piping run of Figure ~~9-F-4~~1 on the following page. 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.

Refer to the top piping run of Figure ~~9-F-4~~1 on the following page. 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.

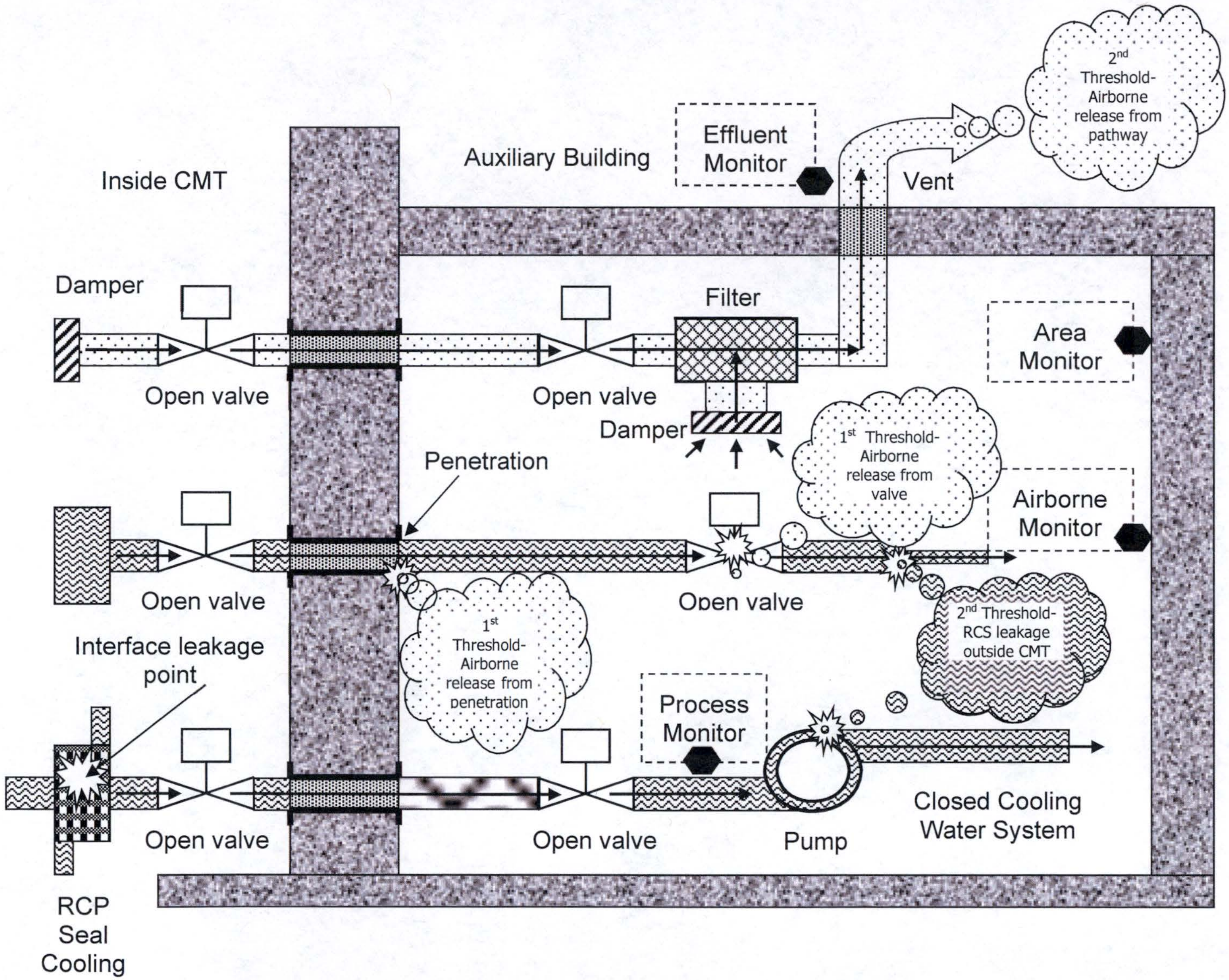
Refer to the bottom piping run of Figure ~~9-F-4~~1 on the following page. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump developed a leak that allowed steam/water to enter the Auxiliary Building, then the second threshold~~ 4.B~~ would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause the first threshold ~~4.A.1~~ to be met as well.

Following the leakage of RCS mass into containment and 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.

DCPP Basis Reference(s):

1. NEI 99-01 CMT Integrity or Bypass Containment Loss 4.A

Figure 1: Containment Integrity or Bypass Examples



Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

2. Indications of RCS LEAKAGE outside of containment.

Definition(s):

RCS LEAKAGE – RCS leakage shall be:

a. Identified Leakage

1. Leakage, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leak off), that is captured and conducted to collection systems or a sump or collecting tank;
2. 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;
3. Reactor Coolant System (RCS) leakage through a steam generator to the secondary system (primary to secondary leakage).

b. Unidentified Leakage

All leakage (except RCP seal water injection or leak off) that is not identified leakage.

c. Pressure Boundary Leakage

Leakage (except primary to secondary leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

d. 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).

Basis:

ERO Decision Making Information

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
- PZR/RCS Loop sample lines

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.

The ECLs resulting from primary leakage outside containment (without a Fuel Clad challenge) are summarized below.

RCS LEAKAGE Outside Containment	ECL
< 25 gpm	No ECL
≥ 25 gpm – Charging Pump capacity	SU5.1
≥ Charging pump capacity	Site Area Emergency based on: RCS Potential Loss A.1 + Containment Loss D.2

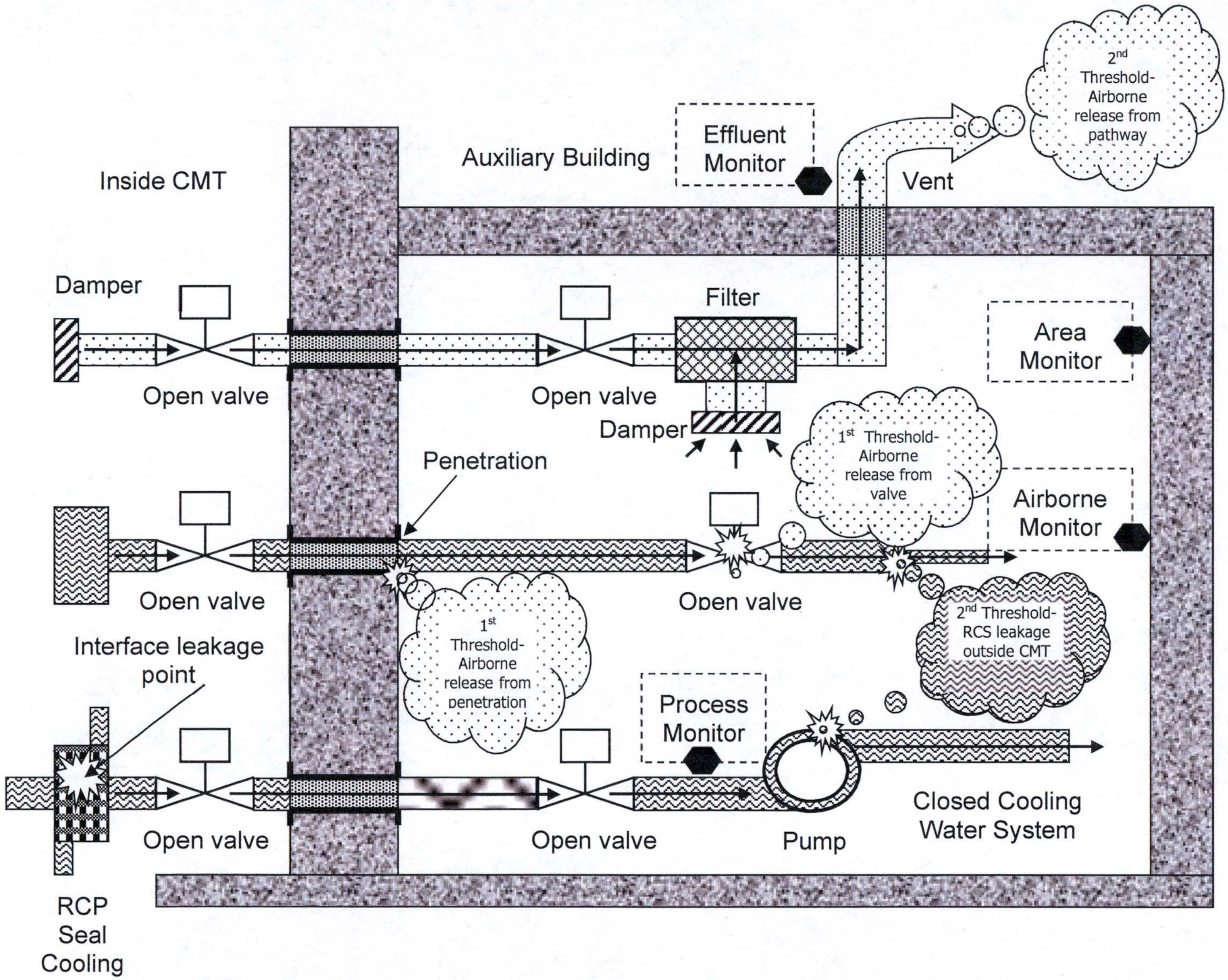
Background

Refer to the middle piping run of Figure ~~9-F-4~~1 on the following page. 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.

DCPP Basis Reference(s):

1. ECA-1.2 LOCA Outside Containment
2. 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. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

1. CSFST Containment - RED path conditions met (≥ 47 psig).

Definition(s):

None

Basis:

ERO Decision Making Information

If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. As noted in the WOG SAMG and related DCPP implementation documents, t~~T~~o 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.

Background

Critical Safety Function Status Tree (CSFST) Containment-RED path is entered if containment pressure is greater than or equal to 47 psig and represents an extreme challenge to safety function. The CSFSTs are normally monitored using the dedicated SPDS display system. Some of the data is also available on the PPC, but the PPC is for information only (ref. 1).

Forty-seven psig is the containment design pressure (ref. 1, 2) and is the pressure used to define CSFST Containment RED path conditions.

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees – Containment, Attachment 5
2. FSAR Appendix 6.2D
3. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.A

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

2. Containment hydrogen concentration ≥ 4%.

Definition(s):

None

Basis:

ERO Decision Making Information

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~~ flammability 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.

Background

After a LOCA, the containment atmosphere is a homogeneous mixture of steam, air, solid and gaseous fission products, hydrogen, and water droplets. During and following a LOCA, the hydrogen concentration in the containment results from radiolytic decomposition of water and metal-water reaction. If hydrogen concentration exceeds the lower flammability limit (4%) in an oxygen rich environment, a potentially explosive mixture exists. Operation of the Containment Hydrogen Recombiner with containment hydrogen concentrations greater than 4% could result in ignition of the hydrogen. If the combustible mixture ignites inside containment, loss of the containment barrier could occur. To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers must also have occurred. Since this threshold is also indicative of loss of both Fuel Clad and RCS barriers with the Potential Loss of the containment barrier, it therefore will likely warrant declaration of a General Emergency (ref. 1, 2, 3, 4).

Containment hydrogen concentration is indicated in the Control Room on ANR-82/ANR-83 PAM1, (range: 1 - 10%).

DCPP Basis Reference(s):

1. UFSAR Section 6.2.5 Combustible Gas Control In Containment
2. FR-Z.4 Response to High Containment Hydrogen Concentration
3. OP-H-9 INSIDE CONT H2 RECOMB SYSTEM
4. CA-3 Hydrogen Flammability in Containment
5. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.B

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

3. Containment pressure ≥ 22 psig.

   AND

   Less than one full train of containment depressurization equipment operating per design for ≥ 15 minutes. (Note 1, 9)

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

Note 9: One Containment Spray pump and two CFCUs comprise one full train of depressurization equipment.

Definition(s):

None

Basis:

ERO Decision Making Information

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 or restore equipment that may not have automatically started or actuated as required, if possible. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems (e.g., containment sprays~~, ice condenser fans~~, etc., but not including containment venting strategies) are either lost or performing in a degraded manner.

Background

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases requirement. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. The Refueling Water Storage Tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, Containment Spray, if needed, is transferred to the RHR Pumps and the Containment Spray Pumps are shut down (ref. 5).

The Containment Cooling System consists of two trains of Containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement. Each train, consisting of two Containment Fan Cooling Units (CFCU), is supplied with cooling water from a separate loop of Component Cooling Water (CCW). In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically if not already running (ref.5).

The Containment pressure setpoint (22 psig, ref. 1, 2, 3, 4) is the pressure at which the equipment should actuate and begin performing its function. The design basis accident analyses and evaluations assume the loss of one Containment Spray System train and one Containment Cooling System train (ref. 5). Consistent with the design requirement, "one full train of depressurization equipment" is therefore defined to be the availability of one train of each system. If less than this equipment is operating and Containment pressure is above the actuation setpoint, the threshold is met if the required equipment cannot be started within 15 minutes.

DCPP Basis Reference(s):

1. AR PK01-18, CONTMT SPRAY ACTUATION red
2. F-0 Critical Safety Function Status Trees - Attachment 6, Containment
3. FR-Z.1 Response to High Containment Pressure
4. Technical Specifications Table 3.3.2-1
5. Technical Specifications B3.6.6 Containment Spray and Cooling Systems
6. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.C

Barrier: Containment

Category: E. SM/SEC/ED Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the SM/SEC/ED that indicates loss of the containment barrier.

Definition(s):

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

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:

ERO Decision Making Information

The SM/SEC/ED 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 SM/SEC/ED 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 SM/SEC/ED~~Emergency Director~~ in determining whether the Containment Barrier is lost.

Background

None

DCPP Basis Reference(s):

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

Barrier: Containment

Category: E. SM/SEC/ED Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the SM/SEC/ED that indicates potential loss of the containment barrier.

Definition(s):

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

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:

ERO Decision Making Information

The SM/SEC/ED 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 SM/SEC/ED 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 SM/SEC/ED~~Emergency Director~~ in determining whether the Containment Barrier is lost.

Background

None

DCPP 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.*

Analysis

OP L-4, Normal Operation at Power (rev 89/73) was reviewed to determine if any actions are "necessary" to **maintain power operations**. Over reasonable periods of time (days vice months or years) there are no actions outside the Control Room that are required to be performed to maintain normal operations. Eventually, you would have to shut down if Technical Specification surveillance testing was not completed and you complied with the associated LCOs or based on consumable supplies being depleted. For the purpose of this table, no actions were determined to be required.

The following table lists the locations into which an operator may be dispatched in order **perform a normal plant cool down and shutdown**. The review was completed using the following procedures as the controlling documents:

OP L-4, Normal Operation at Power (rev 89/73) –

- Sections 6.3 (Instructions for Power Decreases from 100% to 50%)
- Section 6.4 (Instructions for Power Reduction From 50% to 20%)

OP L-5, Plant Cooldown From Minimum Load to Cold Shutdown (rev 100/83)

OP L-7, Plant Stabilization Following Reactor Trip (rev 24/22)

OP AP-25, Rapid Load Reduction or Shutdown (rev 25/12)

In addition, the Residual Heat Removal System is aligned per OP B-2:V "RHR - Place In Service" (rev 37/36) which was also used to conduct this review.

At DCPP, RCS Cooldown starts at OP L-5 step 6.2.3.m.

Each step in the controlling procedures was evaluated to determine if the action was performed in the Control Room or in the plant. Each in-plant action listed below was evaluated and a determination made whether or not the actions, if not performed, would prevent achieving cold shutdown. The following generic assumptions were applied:

- Steps involving optional degassing of the RCS were not selected since degassing the RCS is not required to reach cold shutdown.
- Steps involving supplying Auxiliary Steam were not selected since AFW can be used to reach cold shutdown if Condenser vacuum is lost.
- Steps involving Main Feed Water Pumps were not selected since AFW can be used to reach cold shutdown if Main Feed Water is not available.
- Steps that are stated as needed when entering an outage are disregarded, as they are optional and not mandatory for placing plant in Cold Shutdown.

Travel paths to the locations where the equipment is operated are not part of the determination of affected room/area, only the rooms/areas where the equipment is actually operated. Most locations can be reached via alternate travel paths if required due to a localized issue.

No assumption is made about which RHR Train is aligned for operation.

The minimum set of in-plant actions, associated locations, and operating modes to shut down and cool down the reactor are highlighted. The locations where those actions are performed comprise the rooms/areas to be included in EAL Tables R-2 and H-2. Specifically, the identified rooms are those where an activity must be performed to borate to cold shutdown, isolate accumulators or cooldown using RHR.

UFSAR Page 6.4-1 states "The DCPP control room, located at elevation 140 feet of the auxiliary building, is common to Unit 1 and Unit 2. The associated habitability systems provide for access and occupancy of the control room during normal operating conditions, radiological emergencies, hazardous chemical emergencies, and fire emergencies."

UFSAR Page 6.4-9 states "There are no offsite or onsite hazardous chemicals that would pose a credible threat to DCPP control room habitability. Therefore, engineered controls for the control room are not required to ensure habitability against a hazardous chemical threat and no amount of assumed unfiltered in-leakage is incorporated into PG&E's hazardous chemical assessment."

Control room habitability relative to area radiation levels is adequately bounded by EAL RA2.3.

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
OP L-4, Section 6.3: Instructions for Power Decreases from 100% to 50% **OP L-4, Section 6.4: Instructions for Power Reduction From 50% to 20%**				
OP L-4 6.3.3.b.2	Initiate RCS degassing as directed by chemistry PER OP B-1A:VIII, "Reactor Coolant System Degassing During a Plant Shutdown" OR OP B 1A:X, "CVCS - VCT Degassing."	Aux/100/various	1	No
OP L-4 6.3.3.I / 6.3.4.e	IF either cylinder heating steam pressure controller is in "MANUAL," THEN direct Turbine Building Watch to maintain cylinder heating pressure during the ramp PER OP C-3A:I, "Sealing Steam System - Place In Service."	TB/104	1	No
OP L-4 6.3.3.n	As power decreases, direct Nuclear Operators to adjust SGBD flows PER OP D-2:V, "Steam Generator Blowdown System - Place in Service."	TB/119	1	No
OP L-4 6.3.3.r.6 / 6.3.4.n.4	Direct operator in the field to open discharge vent to condenser valve on condensate pump that was just shut down: • CND PP 1-1: CND-1-31 • CND PP 1-2: CND-1-32 • CND PP 1-3: CND-1-33	TB/85	1	No
OP L-4 6.3.3.s / 6.3.4.i	WHEN less than 60% power, AND IF desired, THEN shut down one of the two running Circulating Water Pumps PER OP E-4:III, "Circulating Water System Shutdown and Clearing."	Intake	1	No
OP L-4 6.3.3.t.4 / 6.3.4.h.4	IF shutdown of MFW pump is required, THEN complete shutdown PER OP C-8:III, "Shutdown and Clearing of a Main Feed Water Pump."	TB/85	1	No
OP L-4 6.3.4.j	IF condenser is to be cleared upon reaching MODE 3, THEN consider realigning TDAFWP steam traps PER appropriate steps in OP L-5, "Plant Cooldown from Minimum Load to Cold Shutdown," section for "Secondary Plant Shutdown."	TB/104 AB/100/Pen	1	No

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
OP L-4 6.3.4.k	Direct Aux Watch to transfer aux steam supply to U2 PER OP K-5:IV "Auxiliary Steam System - Change Over to Alternate Supply of Steam."	AB/140	1	No
OP L-4 6.3.4.l	IF NOT already performed, THEN swap the Hydrazine injection points to the alternate alignment (downstream of FCV-232) per OP D-2:II, "Main Feed Water Chemical Injection System - Place in Service."	TB/85	1	No
OP L-4 6.3.4.m.2.a	IF unit is NOT being taken off line for OP L-8, "Separating From the Grid While Maintaining Reactor Power Between 17% and 30%"), THEN shut down No. 2 Heater Drip Pump PER OP C-7B:II, "No. 2 Heater Drip Pump Shutdown and Clearing."	TB/104, 85 & 70	1	No
OP L-4 6.3.4.t	On the MFW pump in service, locally place the HP and LP Stop Valves Drain control switch to the "OPEN" position to open the before-seat drains.	TB/85	1	No
OP L-5 Section 6.1.3: Power Decrease from 20% to MODE 3 with Normal Shutdown **OP L-5 Section 6.1.4: Power Decrease from 20% to MODE 3 with Planned Reactor Trip**				
OP L-5 6.1.3.d.2	IF Containment is to be entered, THEN Notify Chemistry to perform Containment air sampling.	Pen/100	1	No
OP L-5 6.1.3.m.13 / 6.1.4.t	Place AFW chemical injection in service PER OP D-2:I, "Auxiliary Feed Water Chemical Injection System - Place In Service."	AB/100	1	No
OP L-5 6.1.3.q	IMPLEMENT Section 10 to open FW-1-FCV-420 to prevent the FWH outlet relief from lifting.	TB	1/2/3	See step by step analysis of Section 10
OP L-5 6.1.3.s	IMPLEMENT step 11.5 for secondary shutdown.	TB	1/2/3	See step by step analysis of Section 11
OP L-5 6.1.3.w.8 / 6.1.4.u	Shut down both MFW pumps PER OP C-8:III, "Shutdown and Clearing of a Main Feed Water Pump."	TB	2/3	No
OP L-5 6.1.3.y.5 / 6.1.4.w	IF desired, THEN shut down the MG sets PER OP A-3:III, "Control Rod System - Shutdown & Clearing."	Area H/100	3	No

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
OP L-5 6.1.3.aa / 6.1.4.e.1	Initiate boration to the final concentration for the mode to which the plant is to be shut down PER one of the following • (Preferred) OP B-1A:XIX, "CVCS - Borate the RCS to Refueling Concentration" • (Alternate) OP B-1A:VII, Section 6.12, "Emergency Boration using CVCS-1-8104" • (Alternate) OP B-1A:VII, Section 6.3, "Routine Boration"	AB 100/115/East End (BAST & BA Pumps), 85' (Aux Control Board), 64' (BART Tank area)	2/3	Yes – basis of location is the ability to refill the BAST in order to have sufficient boric acid to reach CSD concentration. Cool down below 500°F requires 11000 gallons of boric acid be added (see step 6.2.3.d)
OP L-5 6.1.3.bb / 6.1.4.x	IF anticipated that the RCS will be opened and degassing of the RCS has not been started, THEN initiate degassing of the RCS to reduce H_2 concentration to 5 cc/kg or less PER OP B-1A:VIII, "CVCS - Reactor Coolant System Degassing During a Plant Shutdown."	AB/100	2/3	No
OP L-5 6.1.3.ee / 6.1.4.z	Maintenance to perform STP M-17B2, "Functional Test of Emergency DC Lighting System in Containment."	Various	1/2/3	No
OP L-5 6.1.3.gg / 6.1.4.bb	Ensure SGBD is maximized PER Chemistry direction and within the ability to control RCS temperature.	TB/119	1/2/3	No
OP L-5 section 6.2: MODE 3 to Ready for RHR Operation				
OP L-5 6.2.3.	Place the personnel airlock automatic leak rate monitor (ALRM) in manual PER STP M-8F1, "ALRM Leak Rate Testing of Personnel Air Lock Seals," to avoid nuisance alarms in the Control Room.	AB/140	3	No
OP L-5 6.2.3.e.2	Borate the RCS to meet STP R-19 COLD SHUTDOWN requirements.	AB 100/115/East End (BAST & BA Pumps), 85' (Aux Control Board), 64' (BART Tank area)	3/4	Yes – basis of location is the ability to refill the BAST in order to have sufficient boric acid to reach CSD concentration. (See Caution prior to step). TS 3.1.1
OP L-5 6.2.3.s	Close the accumulator isolation valve breakers • SI-1-8808A: breaker 52-1F-46 • SI-1-8808B: breaker 52-1G-07 • SI-1-8808C: breaker 52-1H-14	Area H/100/480V Buses	3/4	Yes – basis is that without closing Accumulator outlet valves, RCS pressure cannot go below ~650

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
	• SI-1-8808D: breaker 52-1G-05			psig (procedure does not address alternate actions) TS 3.5.1
OP L-5 6.2.3.y	WHEN desired, THEN secure the CRDM fans PER OP H-6:II, "CRDM Fans - Shutdown and Clearing."	Area H/100/480V Buses	3/4	No
OP L-5 6.2.3.cc.1.b	Disable BOTH SI Pumps PER OP O-32,"Control of Refueling Tags."	TB/119/4kV Vital Bus Rooms	3/4	No – ECG 8.6 violation, but does not prevent getting to CSD
OP L-5 6.2.3.cc.2.b	Disable ONE ECCS centrifugal charging pump PER OP O-32, "Control of Refueling Tags."	TB/119/4kV Vital Bus Rooms	3/4	No – ECG 8.6 violation, but does not prevent getting to CSD
OP L-5 section 6.3: Placing RHR in Service to CSD, Bubble in PZR				
OP L-5 6.3.3.b.4	Place RHR system in service PER OP B-2:V, "RHR-Place in Service During Plant Cooldown."		3/4	See step by step analysis of OP B-2:V
OP L-5 6.3.3.b.6	Place tags on RHR suction valves (RHR-1-8701 and RHR-1-8702) breakers PER OP O-32, "Control of Refueling Tags."	Area H/100/480V Buses	3/4	No – This is only a tag hanging step. Actual breaker manipulation is in OP B-2:V steps 6.2.12 / 6.3.12
OP L-5 6.3.3.d.1	Perform the following actions for CCP 1-3: Establish fire watch compensatory actions per ECG 8.1.	AB/73/CCP3 room	4	No – ECG 8.1 violation, but does not prevent getting to CSD
OP L-5 6.3.3.d.2	Perform the following actions for CCP 1-3: No more than one hour prior to reducing any WR RCS TCOLD to 283°F, make CCP 1-3 incapable of injecting.	TB/119/4kV Vital Bus Rooms	4	No – ECG 8.1 violation, but does not prevent getting to CSD
OP L-5 6.3.3.h	Hang the RCS Dilution Flow Path Boundary valve tags PER OP O-32, "Control of Refueling Tags."	AB/100	4	No
OP L-5 Section 10: Condensate System Long Recirc				
10.2	Ensure CLOSED FW-1-383, FCV-420 Downstream Isolation.	TB/85	3/4	No
10.3	Ensure CLOSED FW-1-384, FCV-420 Downstream Isolation Bypass	TB/85	3/4	No
10.4.1	Open FW-1-210, FW-1-211 Bypass.	TB/85	3/4	No
10.4.2	Open FW-1-211	TB/85	3/4	No
10.4.3	Close FW-1-210	TB/85	3/4	No

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
10.6	Ensure a minimum of four polisher vessels in service until long recirc is established.	TB/85	3/4	No
10.8.2	IF the temperature interlock is NOT made up, THEN contact Maintenance to open FW-1-FCV-420 by installing an air jumper with a 50 psig air supply connected to the vent side of SV1420.	TB/85	3/4	No
10.9	Coordinate with the Control Room and very slowly open FW-1-384 until the onset of FWH flashing, then throttle closed until it stops	TB/85	3/4	No
10.12	Slowly begin to open FW-1-383. If FWH flashing occurs, then throttle closed until it stops.	TB/85	3/4	No
10.14	Close FW-1-384.	TB/85	3/4	No
OP L-5 Section 11: Secondary System Shutdown				
11.2.2.a	Perform the following to prepare steam line drains for closing the MSIVs: Align valves for steam traps 1, 2, 3 and 5 steam line drains.	TB/104	3/4	No – If steam traps cannot be re-aligned declare AFW Pump 1 INOPERABLE.
11.2.2.b	Align AFW Pump 1-1 and Main Steam Traps 1, 2, 3 and 5 to the Outfall	TB/104 & Pen/100	3/4	No – If steam traps cannot be re-aligned declare AFW Pump 1 INOPERABLE.
11.2.3	Connect hoses for AFW chemical injection PER OP D-2:IV, "Adding Chemicals to Chemical Day Tanks-AFW System."	AB/100/AFW room	3/4	No
11.2.5	IF desired, THEN secure and clear a CWP PER OP E-4:III, "Circulating Water System Shutdown and Clearing."	Intake	3/4	No
11.2.7	IF the Main Generator is to be depressurized and purged, THEN warm up the CO2 vaporizer PER OP J-4C:III, "Generator Hydrogen System-Remove From Service."	TB/104	3/4	No
11.3	Just prior to separating from grid, drain MSR drain tanks and FW heaters PER OP C-7:III, "Condensate System - Shutdown and Layup."	TB/119	3/4	No
11.5.2	IF relatching the Main Turbine is	TB/140	3/4	No – If Cooldown

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
	needed to control plant cool down, THEN perform the following:[T35016] a. Close AIR-I-1-2489, Air Supply to the Air/Oil Relay. b. Isolate EH to the governor valves: • EH-1-518, for FCV-139 • EH-1-519, for FCV-140 • EH-1-520, for FCV-141 • EH-1-521, for FCV-142			control is an issue then MSIVs can be closed
11.5.3.b	Align the MSRs as necessary PER OP C-5:III, "Moisture Separator Reheaters - Shutdown."	TB/119 & 104	3/4	No
11.5.5	IF desired, THEN back feed the unit from 500kV PER OP J-2:V, "Back feeding the Unit from the 500kV System."	Various	3/4	No
11.5.7	Secure and drain SCCW PER OP J-4A:III, "Generator Stator Cooling Water-Shutdown and Draining."	TB/85	3/4	No
11.6.1	Depressurize and purge the Main Generator PER OP J-4C:III, "Generator Hydrogen System-Remove from Service."	TB/140 & 119	3/4	No
11.6.2	Secure SCW to exciter air coolers	TB/104	3/4	No
11.7.6	Remove polishers from service PER OP C-7C:II, "Condensate Polishing System-Remove Demineralizers from Service," as directed by the Secondary Foreman.	TB/85 & 104/Polishers	3/4	No
11.7.8	Open CND-1-506 to break vacuum.	TB/119	3/4	No
11.7.9	Maintenance to remove RM-15 and RM-15R from service.	TB/104	3/4	No
11.7.10	Secure gland steam and cylinder heating steam PER OP C-3A:III, "Sealing Steam System-Shutdown and Clearing.	TB/104 & 140	3/4	No
11.7.11	Secure condenser air removal PER OP C-6:III, "Condenser and Air Removal System-Shutdown and Clearing."	TB/104	3/4	No
11.7.12	Secure the following PER OP C-6C:II, "Condensate Air and Nitrogen Injection - Remove from Service:"	TB/119 & 140	3/4	No

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
	• N2 injection • Air injection			
11.7.14.b	Secure chemical injection PER OP D-2:II, "Main Feed Water Chemical Injection-Place in Service."	TB/85	3/4	No
11.7.15	Isolate condensate reject PER OP C-7:III, "Condensate System-Shutdown and Layup" (LCV-12).	TB/85	3/4	No
11.11.1	Secure turning gear PER OP C-3:IV, "Main Unit Turbine-Turbine Shutdown."	TB/140	3/4	No
11.11.2	Shut down lube oil PER OP C-3B:III, "Lube Oil Distribution System-Shutdown and Clearing."	TB/85, 104 & 119	3/4	No
11.11.4	Shut down H2 Seal Oil System PER OP J-4B:II, "Hydrogen Seal Oil System-Shutdown and Drain."	TB/85	3/4	No
11.12	WHEN the RCS is at or below 350°F, THEN remove locking devices on the following SGBD throttle valves and open them to achieve maximum blowdown	TB/119 & AB/140	3/4	No
OP B-2:V: RHR - Place In Service				
6.1.5	Shift chemistry/radiation protection technician to sample RHR Loop 1-1 to determine RHR Loop 1-1 boron concentration.	AB/100/PSSS	4	No - Basis is the system is aligned for ECCS Mode from the RWST. In the event of an SI, we do not verify boron concentration prior to injecting.
6.1.10	Shift chemistry/radiation protection technician to sample RHR Loop 1-2 to determine RHR Loop 1-2 boron concentration.	AB/100/PSSS	4	No - Basis is the system is aligned for ECCS Mode from the RWST. In the event of an SI, we do not verify boron concentration prior to injecting.
6.1.13.b / 6.2.27 / 6.2.43	Open RHR-1-8734A, RHR System 1-1 Bypass to Letdown Heat Exchanger Inlet (85' Containment Penetration Area).	Pen/85	4	No - Basis is the system is aligned for ECCS Mode from the RWST. In the event of an SI, we do not verify boron concentration prior to injecting.
6.1.13.i	Chemistry to sample RHR Loop 1-1 at approximately 10 minute	AB /100/PSSS	4	No - Basis is the system is aligned for ECCS

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
	intervals until the boron concentration is equal to or greater than that in the RCS.			Mode from the RWST. In the event of an SI, we do not verify boron concentration prior to injecting.
6.1.13.l / 6.2.36.b	Close RHR-1-8734A, RHR System 1-1 Bypass to Letdown Heat Exchanger Inlet.	Pen/85	4	No
6.1.13.q / 6.2.36.a / 6.3.8	Open RHR-1-8734B, RHR System 1-2 Bypass to Letdown Heat Exchanger Inlet (85'Containment Penetration Area).	Pen/85	4	No - Basis is the system is aligned for ECCS Mode from the RWST. In the event of an SI, we do not verify boron concentration prior to injecting.
6.1.13.u	Chemistry to sample the RHR loop at approximately 10 minute intervals until the boron concentration of RHR loop 1-2 is equal to or greater than that in the RCS.	AB /100/PSSS	4	No - Basis is the system is aligned for ECCS Mode from the RWST. In the event of an SI, we do not verify boron concentration prior to injecting.
6.1.13.w / 6.2.18	Close RHR-1-8734B, RHR System 1-2 Bypass to Letdown Heat Exchanger Inlet.	Pen/85	4	No
6.2.9 / 6.3.9	Open RHR-1-8726A, RHR Heat Exchanger 1-1 Bypass (64' elevation Auxiliary Building).	AB/64/RHR pumps hallway	4	No – This keeps the RHR trains split but does not prevent cool down.
6.2.10 / 6.3.10	Open RHR-1-8726B, RHR Heat Exchanger 1-2 Bypass (64' elevation Auxiliary Building).	AB/64/RHR pumps hallway	4	No – This keeps the RHR trains split but does not prevent cool down.
6.2.12 / 6.3.12	Ensure CLOSED the breakers for the following valves: • 52-1F-31, MOV 8980 • 52-1G-25, MOV 8701 • 52-1H-19, MOV 8702	Area H/100/480V Buses	4	Yes – required to align RHR system
OP L-7, Plant Stabilization Following Reactor Trip				
6.5.2	IF a Circulating Water pump was tripped, THEN REFER TO OP E-4:III, Circulating Water System – Shutdown and Clearing, for cleanup actions.	Intake	3	No
6.5.3	IF no Circulating Water pump can be placed in service, THEN cool	TB/various	3	No

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
	down a hot condenser in accordance with AP-7, Attachment 1			
6.10.7	Align SG Blowdown via the Blowdown Tank per OP D-2:V for SG chemistry and RCS temperature control.	TB/119 & Pen/100	3	No
6.11.4	Condensate Polisher Beds aligned per Secondary Foreman direction.	TB/104/Polisher	3	No
6.12.2.i	Open FW-1-FCV-420	TB/104	3	No
6.12.2.j	Coordinate with the Control Room and very slowly OPEN FW-1-384	TB/85	3	No
6.12.2.k	Very slowly OPEN FW-1-383.	TB/85	3	No
6.12.2.l	Close FW-1-384	TB/85	3	No
6.13.2.b	Realign steam traps 1, 2, 3, and 5 / steam line drains	TB/104	3/4	No – If steam traps cannot be re-aligned declare AFW Pump 1 INOPERABLE.
6.13.2.c & d	Align AFW Pump 1-1 and Main Steam Traps 1, 2, 3 and 5 to the Outfall	TB/104 & Pen/100	3/4	No – If steam traps cannot be re-aligned declare AFW Pump 1 INOPERABLE.
6.13.3	Align Auxiliary and Gland Seal steam as desired per OP C-3A:I.	TB/104 & AB/100	3/4	No
6.14.2	If desired to control plant cool down, relatch the Main Turbine as follows: a. Close AIR-I-1-2489, Air Supply to the Air/Oil Relay. b. Isolate EH to the governor valves: • EH-1-518, for FCV-139 • EH-1-519, for FCV-140 • EH-1-520, for FCV-141 • EH-1-521, for FCV-142	TB/140	3/4	No – If cooldown control is an issue then MSIVs can be closed
6.15	IF desired, THEN back feed the unit from 500kV PER OP J-2:V, "Back feeding the Unit from the 500kV System."	Various	3/4	No
6.31	On the 4kV vital buses, reset dropped flags on undervoltage relays 27HFB1, 27HGB1 and 27HHB1.	TB/119/4kV Vital Bus Rooms	3/4	No
OP AP-25, Rapid Load Reduction or Shutdown				

ATTACHMENT 3
Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
7.a RNO e / 14.f.4 / 20.c.4	WHEN plant conditions permit, THEN swap Condensate Pump vents PER OP C-7A:I.	TB/85	1/2/3	No

Table R-2 & H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode(s)**
Auxiliary Building – 115' - BASTs	2, 3, 4
Auxiliary Building – 100' – BA Pumps	2, 3, 4
Auxiliary Building – 85' – Aux Control Board	2, 3, 4
Auxiliary Building – 64' – BART Tank area	2, 3, 4
Area H (below Control Room) – 100' 480V Bus area/rooms	3, 4

EAL Technical Basis Document, Revised

Appendix D - Emergency Action Level Technical Basis Document

9/27/16

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 Diablo Canyon Power Plant (DCPP). It should be used to facilitate review of the DCPP EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EP G-1 Emergency Classification and Emergency Plan Activation, may use this document as a technical reference in support of EAL interpretation. This information may assist the SM/SEC/ED 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.

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

2.0 DISCUSSION

2.1 Background

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

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

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

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

Subsequently, Revision 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 (ref. 4.1.1), DCPP 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 (CMT): 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

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 DCPP EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
  - EALs applicable under <u>any</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, 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 DCPP EAL categories are aligned to and represent the NEI 99-01"Recognition Categories." Subcategories are used in the DCPP scheme as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The DCPP EAL categories and subcategories are listed below.

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
R – Abnormal **R**ad Levels / Rad Effluent	1 – Radiological Effluent 2 – 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 – SM/SEC/ED 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 – RTS 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 Wall Chart. The user of the EAL Classification Wall Chart 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 (Any, Hot, Cold), EAL category (R, C, H, S, E and F) and EAL subcategory. A summary explanation of each category and subcategory is given at the beginning of the technical bases discussions of the EALs included in the category. For each EAL, the following information is provided:

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:

An EAL basis section that provides both generic and site-specific ERO decision making guidance as well as background information that supports the rationale for the EAL as provided in NEI 99-01 Rev. 6.

DCPP 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 °F with all reactor vessel head closure bolts fully tensioned

5 Cold Shutdown

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

6 Refueling

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

D Defueled

Reactor vessel contains no irradiated fuel (full core off-load during refueling or extended outage).

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

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Shift Manager/Site Emergency Coordinator/Emergency Director (SM/SEC/ED) 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 (ref. 4.1.9).

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 SM/SEC/ED should not wait until the applicable time has elapsed. The SM/SEC/ED 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 cannot be determined, 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). For example, a coolant activity sample is taken. Chemistry reports results indicate activity greater than Technical Specification limits. The classification clock begins when Chemistry reports the sample results.

3.1.6 SM/SEC/ED 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 SM/SEC/ED 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 SM/SEC/ED 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." (ref. 4.1.9).

3.2.1 Classification of Multiple Events and Conditions

When multiple emergency events or conditions are present, the user will recognize 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.

For example, a loss of decay heat removal when in Mode 5 results in RCS temperature exceeding 200ºF. Escalation of the loss of decay heat removal event will be via the cold condition mode EALs even though the plant is now in Mode 4 as a result of the RCS temperature increase. However, any subsequent new event/condition must be assessed against the hot condition EALs (Mode 4 and above).

3.2.3 Classification of Imminent Conditions

Although EALs provide specific thresholds, the SM/SEC/ED 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 SM/SEC/ED, 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. Refer to EP G-1 Emergency Classification and Emergency Plan Activation for guidance on downgrading and terminating an ECL. Refer to EP OR-3 Emergency Recovery for guidance for entering long-term recovery.

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.

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

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

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

It is important to stress that the 15-minute emergency classification assessment period (process clock) is not a "grace period" during which a classification may be delayed to allow the performance of a corrective action that would preclude 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 SM/SEC/ED completing the review and steps necessary to make the emergency declaration. This provision is included to ensure that any public protective actions resulting from the emergency classification are truly warranted by the plant conditions.

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

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

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 (ref. 4.1.3) is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR § 50.72 (ref. 4.1.4) within one hour of the discovery of the undeclared event or condition (refer to XI1.ID2 Regulatory Reporting Requirements and

Reporting Process (ref. 4.1.11)). 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 10CFR 50.72 Immediate Notification Requirements for Operating Nuclear Power Reactors

4.1.5 10CFR 50.73 License Event Report System

4.1.6 Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points

4.1.7 Technical Specifications Table 1.1-1 Modes

4.1.8 Administrative Procedure AD8.DC54 "Containment Closure"

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

4.1.10 DCPP Emergency Plan

4.1.11 XI1.ID2 Regulatory Reporting Requirements and Reporting Process

4.1.12 DCPP Security and Safeguards Contingency Plan

4.2 Implementing

4.2.1 EP G-1 Emergency Classification and Emergency Plan Activation

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

4.2.3 DCPP EAL Wall Chart

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions

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 process, 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.

Confinement Boundary

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

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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 "Containment Closure" (ref. 4.1.8).

Degraded Performance

As applied to hazardous event thresholds, damage significant enough to cause concern regarding the operability or reliability of the affected safety system train.

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 Protective Action Guidelines (EPA PAG)

The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs requires DCPP 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. (*refer to Section 2.2*)

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 process 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 DCPP 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 DCPP. 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.

ISFSI Protected Area

Areas within the ISFSI to which access is strictly controlled in accordance with the station's Security Plan.

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) (ref. 4.1.8).

Owner Controlled Area (OCA)

For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan (ref. 4.1.12).

Projectile

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

Plant Protected Area

Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the Plant Protected Area.

RCS Leakage

RCS Leakage shall be:

a. Identified Leakage
   1. Leakage, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leak off), that is captured and conducted to collection systems or a sump or collecting tank;
   2. 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;
   3. Reactor Coolant System (RCS) leakage through a steam generator to the secondary system (primary to secondary leakage).

b. Unidentified Leakage

   All leakage (except RCP seal water injection or leak off) that is not identified leakage.

c. Pressure Boundary Leakage

   Leakage (except primary to secondary leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

d. 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).

Reduced Inventory Condition (RIC)

The condition existing whenever RCS water level is lower than 3 feet below the reactor vessel flange (below 111-foot elevation) with fuel in the core.

Refueling Pathway

The 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.

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.

Security Event

Any incident representing an attempted, threatened, or actual breach of the security system or reduction of the operational effectiveness of that system. A security event can result in either a SECURITY CONDITION or HOSTILE ACTION (ref. 4.1.12).

Site Area Emergency

Events are in process 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 Boundary

As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points (ref. 4.1.6).

Tornado

A violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

Unisolable

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

NOTE: If an isolation valve could not be accessed due to local conditions (i.e. high radiation, temperature, etc.) then that would also make a leak unisolable, even though the inaccessible valve could isolate the leak.

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 process 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.

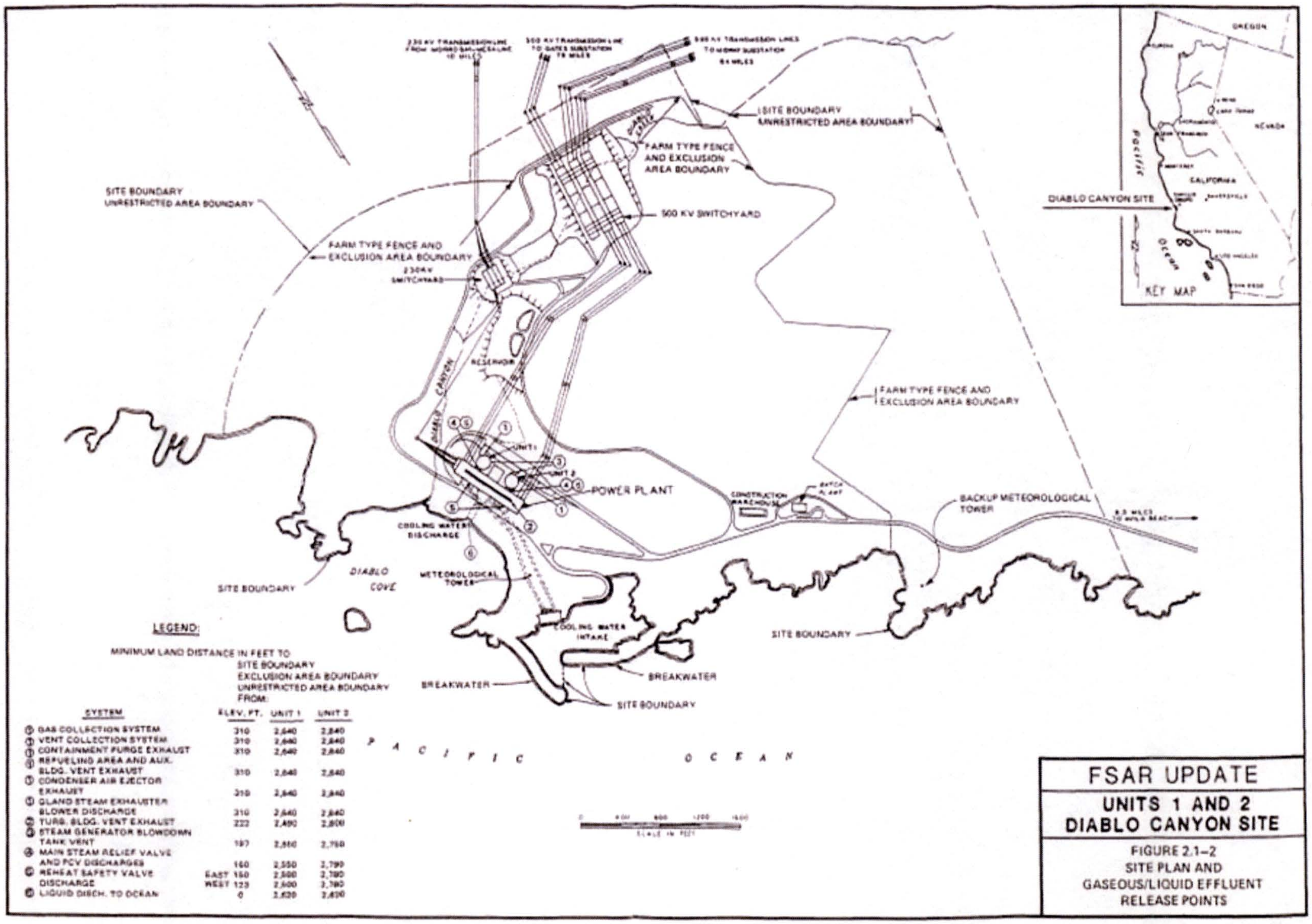
No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

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. 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.



LEGEND:

MINIMUM LAND DISTANCE IN FEET TO
SITE BOUNDARY
EXCLUSION AREA BOUNDARY
UNRESTRICTED AREA BOUNDARY
FROM:

SYSTEM	ELEV. FT.	UNIT 1	UNIT 2
GAS COLLECTION SYSTEM	310	2,640	2,840
VENT COLLECTION SYSTEM	310	2,640	2,840
CONTAINMENT PURGE EXHAUST	310	2,640	2,840
REFUELING AREA AND AUX. BLDG. VENT EXHAUST	310	2,640	2,840
CONDENSER AIR EJECTOR EXHAUST	310	2,640	2,840
GLAND STEAM EXHAUSTER BLOWER DISCHARGE	310	2,640	2,840
TURB. BLDG. VENT EXHAUST	222	2,490	2,800
STEAM GENERATOR BLOWDOWN TANK VENT	197	2,850	2,750
MAIN STEAM RELIEF VALVE AND PCV DISCHARGES	160	2,550	2,790
REHEAT SAFETY VALVE DISCHARGE	EAST 150	2,500	2,780
	WEST 123	2,500	2,780
LIQUID DISCH. TO OCEAN	0	2,620	2,620

FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 2.1-2
SITE PLAN AND
GASEOUS/LIQUID EFFLUENT
RELEASE POINTS

Revision 11 November 1996

5.2 Abbreviations/Acronyms

°F	Degrees Fahrenheit
°	Degrees
AC	Alternating Current
AFW	Auxiliary Feedwater
AOP	Abnormal Operating Procedure
ATL	Automatic Tie Line
ATWS	Anticipated Transient Without Scram
BA	Boric Acid
BART	Boric Acid Reserve Tank
BAST	Boric Acid Storage Tank
CAS	Central Alarm Station
CCW	Component Cooling Water
CC(#)	Control Console (number)
CDE	Committed Dose Equivalent
CEDE	Committed Effective Dose Equivalent
CET	Core Exit Thermocouple
CFCU	Containment Fan Cooling Unit
CFR	Code of Federal Regulations
CMT	Containment
CSFST	Critical Safety Function Status Tree
CST	Condensate Storage Tank
DBA	Design Basis Accident
DC	Direct Current
DCPP	Diablo Canyon Power Plant
DDE	Double Design Earthquake
DE	Design Earthquake
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
ECL	Emergency Classification Level
ED	Emergency Director
EDE	Effective Dose Equivalent
EFM	Earthquake Force Monitor
ENF	Emergency Notification Form
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

ERFDS .... Emergency Response Facility Display System
FAA .... Federal Aviation Administration
FBI .... Federal Bureau of Investigation
FEMA .... Federal Emergency Management Agency
FSAR .... Final Safety Analysis Report
ft. .... Feet
FTS .... Federal Telephone System
GDC .... General Design Criteria
GE .... General Emergency
HASP .... High Alarm Setpoint
HOO .... NRC Headquarters Operations Officer
IC .... Initiating Condition
in .... Inches
IPEEE .... Individual Plant Examination of External Events (Generic Letter 88-20)
K_{eff} .... Effective Neutron Multiplication Factor
LCO .... Limiting Condition of Operation
LER .... Licensee Event Report
LOCA .... Loss of Coolant Accident
LWR .... Light Water Reactor
MEDT .... Miscellaneous Equipment Drain Tank
MPC .... Maximum Permissible Concentration
MPC .... Multi-Purpose Canister
mR, mRem, mrem, mREM .... milli-Roentgen Equivalent Man
MSL .... Main Steam Line
MW .... Megawatt
NEI .... Nuclear Energy Institute
NEIC .... National Earthquake Information Center
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
NUREG .... Nuclear Regulation
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM .... Off-site Dose Calculation Manual
OES .... Office of Emergency Services
ORO .... Offsite Response Organization
PA .... Protected Area

PAG .... Protective Action Guideline
PAM .... Post Accident Monitoring
PAR .... Protective Action Recommendation
PBX .... Private Branch Exchange
PGA .... Peak Ground Acceleration
PPC .... Plant Process Computer
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PRT .... Pressurizer Relief Tank
PSIG .... Pounds per Square Inch Gauge
PTS .... Pressurized Thermal Shock
PWR .... Pressurized Water Reactor
R .... Roentgen
RCC .... Reactor Control Console
RCDT .... Reactor Coolant Drain Tank
RCS .... Reactor Coolant System
RHR .... Residual Heat Removal
Rem, rem, REM .... Roentgen Equivalent Man
RETS .... Radiological Effluent Technical Specifications
RTS .... Reactor Trip System
R(P)V .... Reactor (Pressure) Vessel
RVLIS .... Reactor Vessel Level Indicating System
RVRLIS .... Reactor Vessel Refueling Level Indicating System
RWST .... Refueling Water Storage Tank
SAE .... Site Area Emergency
SAMG .... Sever Accident Management Guideline
SAR .... Safety Analysis Report
SAS .... Secondary Alarm Station
SBO .... Station Blackout
SCBA .... Self-Contained Breathing Apparatus
SCMM .... Sub Cooled Margin Monitor
SEC .... Site Emergency Coordinator
SG .... Steam Generator
SI .... Safety Injection
SM .... Shift Manager
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
SSF .... Safe Shutdown Facility
TC .... Thermocouple
TEDE .... Total Effective Dose Equivalent
TOAF .... Top of Active Fuel

TSC .... Technical Support Center
UE .... Unusual Event
UFSAR .... Updated Final Safety Analysis Report
USGS .... United States Geological Survey
VB(#) .... Vertical Board (number)
VDC .... Volts Direct Current
WOG .... Westinghouse Owners Group
WR .... Wide Range
XFMR .... Transformer

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

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

DCPP	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

DCPP	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	1
CS1.2	CS1	2
CS1.3	CS1	3
CG1.1	CG1	1
CG1.2	CG	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

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

DCPP	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
SA9.1	SA9	1
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1
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

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.

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.1 Unusual Event

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

Note 1: The SM/SEC/ED 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 **cannot** be determined, 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	Plant Vent	1(2)-RM-14/14R	-----	2.5E+6 cpm	2.5E+5 cpm	8.0E+4 cpm
				5.6E-2 μCi/cc	5.6E-3 μCi/cc	1.8E-3 μCi/cc
		1(2)-RM-87	1.9E-10 amps	----	----	----
			3.2E-1 μCi/cc			
Liquid	Liquid Radwaste Effluent Line	0-RM-18	-----	-----	-----	1.6E+5 cpm
	SGBD Tank	1(2)-RM-23	-----	-----	-----	2.0E+4 cpm

Mode Applicability:

All

Definition(s):

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

Basis:

ERO Decision Making Information

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.

Even if a release does not meet the levels of this EAL, a release may be reportable. In these cases, consult Admin Procedure XI1.ID2.

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.

Background

The column "UE" gaseous and liquid release values in Table R-1 represent two times the appropriate Offsite Dose Calculation Manual 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.

DCPP Basis Reference(s):

1. DCPP Radiological Effluent Technical Specifications
2. EP-CALC-DCPP-1601 Radiological Effluent EAL Values
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 Offsite Dose Calculation Manual 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 Offsite Dose Calculation Manual limits for ≥ 60 minutes. (Notes 1, 2)	

Note 1: The SM/SEC/ED 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 **cannot** be determined, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

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

NOTE: If an isolation valve could not be accessed due to local conditions (i.e. high radiation, temperature, etc.) then that would also make a leak unisolable, even though the inaccessible valve could isolate the leak.

Basis:

ERO Decision Making Information

This EAL addresses uncontrolled gaseous or liquid releases that are detected by sample analyses or environmental surveys (particularly on unmonitored and/or UNISOLABLE pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leaks into river water systems, etc.).

Sample analysis results relative to Offsite Dose Calculation Manual limits are provided by Chemistry.

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

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

Background

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.

DCPP Basis Reference(s):

1. DCPP Radiological Effluent Technical Specifications
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 minutes. (Notes 1, 2, 3, 4)

Note 1: The SM/SEC/ED 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 **cannot** be determined, 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 **only** be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

<table>
<tr><th colspan="7">Table R-1 Effluent Monitor Classification Thresholds</th></tr>
<tr><th colspan="2">Release Point</th><th>Monitor</th><th>GE</th><th>SAE</th><th>Alert</th><th>UE</th></tr>
<tr><td rowspan="4">Gaseous</td><td rowspan="4">Plant Vent</td><td rowspan="2">1(2)-RM-14/14R</td><td rowspan="2">-----</td><td>2.5E+6 cpm</td><td>2.5E+5 cpm</td><td>8.0E+4 cpm</td></tr>
<tr><td>5.6E-2 μCi/cc</td><td>5.6E-3 μCi/cc</td><td>1.8E-3 μCi/cc</td></tr>
<tr><td rowspan="2">1(2)-RM-87</td><td>1.9E-10 amps</td><td rowspan="2">----</td><td rowspan="2">----</td><td rowspan="2">----</td></tr>
<tr><td>3.2E-1 μCi/cc</td></tr>
<tr><td rowspan="2">Liquid</td><td>Liquid Radwaste Effluent Line</td><td>0-RM-18</td><td>-----</td><td>-----</td><td>-----</td><td>1.6E+5 cpm</td></tr>
<tr><td>SGBD Tank</td><td>1(2)-RM-23</td><td>-----</td><td>-----</td><td>-----</td><td>2.0E+4 cpm</td></tr>
</table>

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

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:

ERO Decision Making Information

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 thyroid CDE

The results from a dose assessment are preferred using actual meteorology, when available. Until available, the pre-calculated effluent monitor values presented in Table R-1 should be used for emergency classification. Once an accurate dose assessment is performed, classification should be based on dose assessment only and not using the effluent monitor values in Table R-1. For example, a Table R-1 Alert effluent threshold is exceeded. However, real-time dose assessment results are available indicating offsite doses less than EAL RA1.2 thresholds. Declaration of an Alert due to EAL RA1.1 is not required.

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.

Background

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.

DCPP Basis Reference(s):

1. EP-CALC-DCPP-1601 Radiological Effluent EAL 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 **only** be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

ERO Decision Making Information

Dose projections are performed by computer-based method (ref. 1, 2, 3). Dose assessments may utilize real-time dose projections and/or field monitoring results.

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.

Background

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.

DCPP Basis Reference(s):

1. EP RB-9, Calculation of Release Rate
2. EP RB-11, Emergency Offsite Dose Calculations
3. EP R-2, Release of Airborne Radioactive Materials Initial Assessment
4. 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 minutes of exposure. (Notes 1, 2)	

Note 1: The SM/SEC/ED 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 **cannot** be determined, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

ERO Decision Making Information

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

Background

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.

DCPP Basis Reference(s):

1. EP R-3 Release of Radioactive Liquids
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 are > 10 mR/hr and are expected to continue for ≥ 60 minutes.
- Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 minutes of inhalation.

(Notes 1, 2)

Note 1: The SM/SEC/ED 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 **cannot** be determined, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

ERO Decision Making Information

EP RB-8, Instructions for Field Monitoring Teams provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

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

Background

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.

DCPP Basis Reference(s):

1. EP RB-8, Instructions for Field Monitoring Teams
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 minutes. (Notes 1, 2, 3, 4)

Note 1: The SM/SEC/ED 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 **cannot** be determined, 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 **only** 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	Plant Vent	1(2)-RM-14/14R	-----	2.5E+6 cpm	2.5E+5 cpm	8.0E+4 cpm
				5.6E-2 μCi/cc	5.6E-3 μCi/cc	1.8E-3 μCi/cc
		1(2)-RM-87	1.9E-10 amps	----	----	----
			3.2E-1 μCi/cc			
Liquid	Liquid Radwaste Effluent Line	0-RM-18	-----	-----	-----	1.6E+5 cpm
	SGBD Tank	1(2)-RM-23	-----	-----	-----	2.0E+4 cpm

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

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:

ERO Decision Making Information

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 thyroid CDE

The results from a dose assessment are preferred using actual meteorology, when available. Until available, the pre-calculated effluent monitor values presented in Table R-1 should be used for emergency classification. Once an accurate dose assessment is performed, classification should be based on dose assessment only and not using the effluent monitor values in Table R-1. For example, a Table R-1 SAE effluent threshold is exceeded. However, real-time dose assessment results are available indicating offsite doses less than EAL RS1.2 thresholds. Declaration of a Site Area Emergency due to EAL RS1.1 is not required.

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.

Background

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 thyroid CDE) (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.

DCPP Basis Reference(s):

1. EP-CALC-DCPP-1601 Radiological Effluent EAL 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 **only** be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

ERO Decision Making Information

Dose projections are performed by computer-based method (ref. 1, 2, 3). Dose assessments may utilize real-time dose projections and/or field monitoring results.

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

Background

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.

DCPP Basis Reference(s):

1. EP RB-9, Calculation of Release Rate
2. EP RB-11, Emergency Offsite Dose Calculations
3. EP R-2, Release of Airborne Radioactive Materials Initial Assessment
4. 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 are > 100 mR/hr and are expected to continue for ≥ 60 minutes.
- Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 minutes of inhalation.

(Notes 1, 2)

Note 1: The SM/SEC/ED 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 **cannot** be determined, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

ERO Decision Making Information

EP RB-8, Instructions for Field Monitoring Teams provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

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

Background

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.

DCPP Basis Reference(s):

1. EP RB-8, Instructions for Field Monitoring Teams
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 minutes. (Notes 1, 2, 3, 4)

Note 1: The SM/SEC/ED 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 **cannot** be determined, 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 **only** 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	Plant Vent	1(2)-RM-14/14R	-----	2.5E+6 cpm	2.5E+5 cpm	8.0E+4 cpm
				5.6E-2 µCi/cc	5.6E-3 µCi/cc	1.8E-3 µCi/cc
		1(2)-RM-87	1.9E-10 amps	----	----	----
			3.2E-1 µCi/cc			
Liquid	Liquid Radwaste Effluent Line	0-RM-18	-----	-----	-----	1.6E+5 cpm
	SGBD Tank	1(2)-RM-23	-----	-----	-----	2.0E+4 cpm

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

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:

ERO Decision Making Information

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 thyroid CDE

The results from a dose assessment are preferred using actual meteorology, when available. Until available, the pre-calculated effluent monitor values presented in Table R-1 should be used for emergency classification. Once an accurate dose assessment is performed, classification should be based on dose assessment only and not using the effluent monitor values in Table R-1. For example, a Table R-1 GE effluent threshold is exceeded. However, real-time dose assessment results are available indicating offsite doses less than EAL RG1.2 thresholds. Declaration of a General Emergency due to EAL RG1.1 is not required.

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.

Background

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 thyroid CDE) (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.

DCPP Basis Reference(s):

1. EP-CALC-DCPP-1601 Radiological Effluent EAL 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 **only** be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

ERO Decision Making Information

Dose projections are performed by computer-based method (ref. 1, 2, 3). Dose assessments may utilized real-time dose projections and/or field monitoring results.

Background

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.

DCPP Basis Reference(s):

1. EP RB-9, Calculation of Release Rate
2. EP RB-11, Emergency Offsite Dose Calculations
3. EP R-2, Release of Airborne Radioactive Materials Initial Assessment
4. 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 are > 1,000 mR/hr and are expected to continue for ≥ 60 minutes.
- Analyses of field survey samples indicate thyroid CDE > 5,000 mrem for 60 minutes of inhalation.

(Notes 1, 2)

Note 1: The SM/SEC/ED 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 **cannot** be determined, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

ERO Decision Making Information

EP RB-8, Instructions for Field Monitoring Teams provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

Background

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.

DCPP Basis Reference(s):

1. EP RB-8, Instructions for Field Monitoring Teams
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 equivalent indication.

AND

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

- RM-58 Spent Fuel Pool Area
- RM-59 New Fuel Area
- RM-2 Containment Area (Mode 6 **only**)
- **Any** temporary installed monitor in vicinity of the REFUELING PATHWAY (when installed)

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:

ERO Decision Making Information

Note that this EAL is applicable only in cases where the elevated reading is due to an UNPLANNED loss of water level.

The Spent Fuel Pool (SFP) low water level alarm setpoint is 23 ft. 9 in. above the top of irradiated fuel seated in the SFP storage racks or 137 feet 4 inches elevation.

The Refueling Cavity low water level alarm setpoint is at 138 feet elevation as measured on Reactor Vessel Refueling Level Indicating System (RVRLIS) (i.e., 24 feet above the top of reactor vessel flange).

The reading on an area radiation monitor (permanently installed or temporary) located near the Reactor Cavity may increase due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Elevated radiation monitor indications to the low alarm setpoint will need to be combined with another indicator (or personnel report) of water loss (ref. 5, 6)

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.

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.

Background

SFP water level at 136 feet 7 inches elevation is the Technical Specification LCO limit (SR 3.7.15) that requires 23 ft. of water above irradiated fuel seated in the Spent Fuel Pool storage racks.

A minimum depth of 23 feet of water over the irradiated fuel assemblies in the SFP and 23 feet of water over the reactor vessel flange in the refueling cavity is maintained to ensure sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits and to ensure that the offsite dose consequences due to a postulated fuel handling accident are acceptable (ref. 1, 2, 3, 4).

Loss of Spent Fuel Pool water inventory results from either a rupture of the pool or transfer canal liner, or failure of the spent fuel cooling system and the subsequent boil-off. Allowing SFP water level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment. The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 feet above the top of the reactor vessel flange.

While a radiation monitor (RM-58, RM-59, RM-2 or temporarily installed monitors in the vicinity of the REFUELING PATHWAY) could detect an increase in dose due to a drop in the water level, it might not be a reliable indication, in and of itself, of whether or not there is adequate shielding from irradiated fuel (ref. 5, 6).

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.

DCPP Basis Reference(s):

1. Technical Specification 3.7.15, SFP Level

2. Technical Specification 3.9.7, Refueling Cavity Water Level
3. AR PK11-04 input 1064, Spent Fuel Pool Lvl/Temp
4. AR PK02-22 input 1185, Rx Vsl Refueling Lvl (red)
5. OP AP-22, Spent Fuel Pool Abnormalities
6. AR PK-11-10, FHB High Radiation
7. 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	**Unusual Event**
Uncovery of irradiated fuel in the REFUELING PATHWAY.	

Mode Applicability:

All

Definition(s):

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

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

Basis:

ERO Decision Making Information

This EAL addresses events that have caused a significant lowering of water level within the REFUELING PATHWAY.

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

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

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

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

Background

These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

This EAL escalates from 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.

DCPP Basis Reference(s):

1. OP AP-21, Irradiated Fuel Damage
2. OP AP-22, Spent Fuel Pool Abnormalities
3. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity.

AND

High alarm on **any** of the following radiation monitors:

- RM-59 New Fuel Storage Area
- RM-58 Spent Fuel Pool Area
- **Any** temporary installed monitor in vicinity of the REFUELING PATHWAY (when installed)
- RM-2 Containment Area (Mode 6 **only**)
- RM-44A/B Containment Ventilation Exhaust (Mode 6 **only**)

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 DCPP ISFSI, the confinement boundary is defined to be the Multi-Purpose Canister (MPC).

Basis:

ERO Decision Making Information

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

This EAL 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 EAL EU1.1.

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

Background

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

The bases for the SFP area radiation high alarms and containment area and ventilation radiation high alarms are a spent fuel handling accident and are, therefore, appropriate for this EAL. In the fuel handling building, a fuel assembly could be dropped in the fuel transfer canal or in the SFP. Should a fuel assembly be dropped in the fuel transfer canal or in the spent fuel pool and release radioactivity above a prescribed level, the area radiation monitors sound an alarm, alerting personnel to the problem. Area radiation monitors in the fuel handling building isolate the normal fuel handling building ventilation system and automatically initiate the recirculation and filtration systems. (ref. 1, 2, 3).

This EAL addresses events that have caused IMMINENT or actual damage to an irradiated fuel assembly.

These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

DCPP Basis Reference(s):

1. OP AP-21, Irradiated Fuel Damage
2. OP AP-22, Spent Fuel Pool Abnormalities
3. I&C RMS Data Book
4. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.3	**Alert**
Lowering of spent fuel pool level to 10 ft. above top of the fuel racks (Level 2).	

Mode Applicability:

All

Definition(s):

None

Basis:

ERO Decision Making Information

This EAL addresses a significant lowering of water level within the spent fuel pool.

For DCPP Plant SFP Level 2 is 10 ft. (plant El. 123' 11") as indicated on LI-801. Backup indication is also available on LI-802. The PPC point for SFP level is L0690A for both units.

Main Annunciator window PK11-04 will alarm at SFP Level 2 (ref. 3).

Escalation of the emergency classification level would be via one or more EALs under IC RS1 or RS2.

Background

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.

Post-Fukushima order EA-12-051 (ref. 1) required the installation of reliable SFP level indication capable of identifying normal minimum level (Level 1 – 134' 5"), SFP level 10 ft. above the top of the fuel racks (Level 2 – 123' 11") and SFP level at the top of the fuel racks (Level 3 – 114' 11").

DCPP Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. PG&E Letter DCL-13-073 Response to Request for Additional Information Regarding Overall Integrated Plan in Response to March 12, 2012, Commission Order to Modify

Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)

3. Procedure AR PK11-04
4. SAP documents 50808058 & 68039896 (Unit 1)
5. SAP documents 50808059 & 68039897 (Unit 2)
6. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RS2.1	**Site Area Emergency**
Lowering of spent fuel pool level to 1 ft. above top of the fuel racks (Level 3).	

Mode Applicability:

All

Definition(s):

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

Basis:

ERO Decision Making Information

This EAL addresses a significant loss of spent fuel pool inventory control leading to IMMINENT fuel damage.

For DCPP Plant SFP Level 3 is 1 ft. (plant El. 114' 11") as indicated on LI-801 (includes 1 ft. instrument uncertainly). Backup indication is also available on LI-802. The PPC point for SFP level is L0690A for both units.

Escalation of the emergency classification level would be via one or more EALs under IC RG1 or RG2.

Background

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 EAL would likely not be met until well after another Site Area Emergency EAL was met; however, it is included to provide classification diversity.

Post-Fukushima order EA-12-051 (ref. 1) required the installation of reliable SFP level indication capable of identifying normal minimum level (Level 1 – 134' 5"), SFP level 10 ft. above the top of the fuel racks (Level 2 – 123' 11") and SFP level at the top of the fuel racks (Level 3 – 114' 11").

DCPP Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. PG&E Letter DCL-13-073 Response to Request for Additional Information Regarding Overall Integrated Plan in Response to March 12, 2012, Commission Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number 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 1 ft. above top of the fuel racks (Level 3) for ≥ 60 minutes. (Note 1)	

Note 1: The SM/SEC/ED 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:

ERO Decision Making Information

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.

For DCPP Plant SFP Level 3 is 1 ft. (plant El. 114' 11") as indicated on LI-801 (includes 1 ft. instrument uncertainly). Backup indication is also available on LI-802. The PPC point for SFP level is L0690A for both units.

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

Background

Post-Fukushima order EA-12-051 (ref. 1) required the installation of reliable SFP level indication capable of identifying normal minimum level (Level 1 – 134' 5"), SFP level 10 ft. above the top of the fuel racks (Level 2 – 123' 11") and SFP level at the top of the fuel racks (Level 3 – 114' 11").

DCPP Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. PG&E Letter DCL-13-073 Response to Request for Additional Information Regarding Overall Integrated Plan in Response to March 12, 2012, Commission Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Order Number 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 **EITHER** of the following areas:

Control Room (0-RM-1 or portable gamma radiation instrument)

OR

Central Alarm Station (by survey)

Mode Applicability:

All

Definition(s):

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

Basis:

ERO Decision Making Information

Areas that meet this threshold include the Control Room and the Central Alarm Station (CAS). 0-RM-1 monitors the Control Room for area radiation (ref. 1). A portable gamma radiation instrument may be installed if 0-RM-1 is out of service. The CAS is included in this EAL because of its' importance to permitting access to areas required to assure safe plant operations.

There is no permanently installed CAS area radiation monitor that may be used to assess this EAL threshold. Therefore this threshold must be assessed via local radiation survey for the CAS (ref. 1). For this EAL the Secondary Alarm Station (SAS) is not considered.

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

Background

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 SM/SEC/ED should consider the cause of the increased radiation levels and determine if another IC may be applicable.

DCPP Basis Reference(s):

1. FSAR Table 11.4-1 Radiation Monitors and Readouts
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(s)**
Auxiliary Building – 115' - BASTs	2, 3, 4
Auxiliary Building – 100' – BA Pumps	2, 3, 4
Auxiliary Building – 85' – Aux Control Board	2, 3, 4
Auxiliary Building – 64' – BART Tank area	2, 3, 4
Area H (below Control Room) – 100' 480V Bus area/rooms	3, 4

Mode Applicability:

2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown

Definition(s):

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

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

Basis:

ERO Decision Making Information

The identified rooms are those where an activity must be performed to borate to cold shutdown, isolate accumulators or cooldown using RHR.

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.

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).

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

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

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

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

Background

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.

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 SM/SEC/ED should consider the cause of the increased radiation levels and determine if another IC may be applicable.

NOTE: EAL RA3.2 mode applicability has been limited to the applicable modes identified in Table R-2 Safe Operation & Shutdown Rooms/Areas. If due to plant operating procedure or plant configuration changes, the applicable plant modes specified in Table R-2 are changed, a corresponding change to Attachment 3 'Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases' and to EAL RA3.2 mode applicability is required."

DCCP Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Rooms/Areas Table 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.

The DCPP ISFSI is located within the OWNER CONTROLLED AREA but outside the PLANT PROTECTED AREA. Therefore SECURITY EVENTS related to the ISFSI are classified under either HU1.1 or HA1.1.

ISFSI PROTECTED AREA - Areas within the ISFSI to which access is strictly controlled in accordance with the station's Security Plan.

OWNER CONTROLLED AREA (OCA) - For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan.

PLANT PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

Category: ISFSI

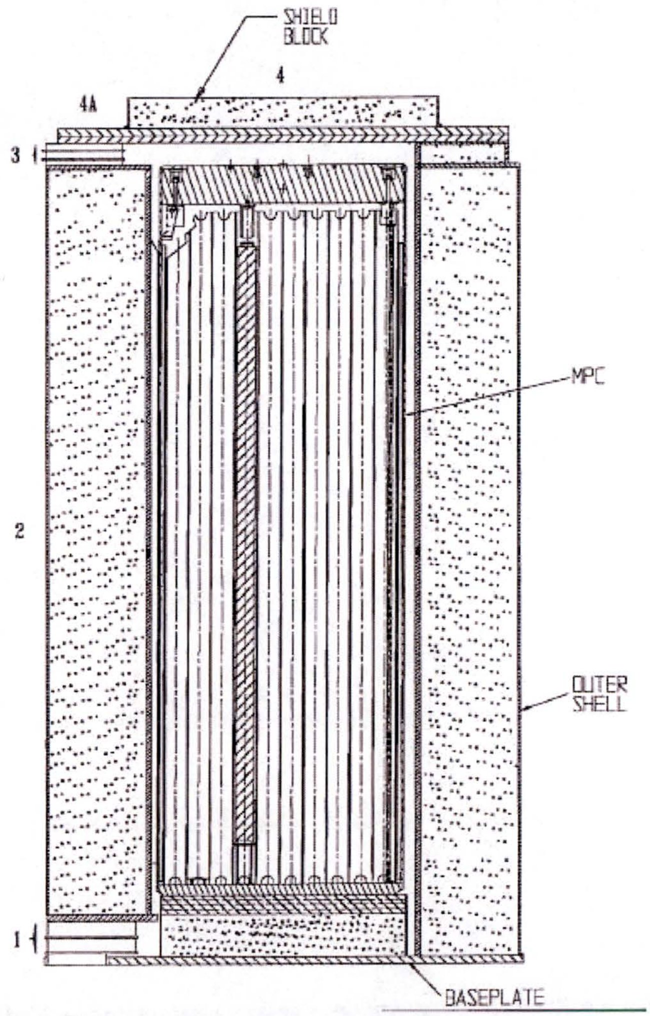
Subcategory: Confinement Boundary

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 > Table E-1.

Table E-1 ISFSI Radiation Readings		
Dose Point Location (see figure)		**Surface Dose Rate** (mRem/hour)
1	Base vent	72
2	Mid plane	80
3	Top vent	76
4	Lid-center	22
4a	Lid-over top vents	139



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 DCPP ISFSI, the confinement boundary is defined to be the Multi-Purpose Canister (MPC).

PLANT PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

Basis:

ERO Decision Making Information

An Unusual Event is declared on the basis of the occurrence of any event of sufficient magnitude that a loaded cask confinement boundary is damaged or violated as indicated by external on-contact dose rates exceeding the maximum calculated levels of an overpack with a loaded MPC-32 canister, based on the locations in the ISFSI FSAR Figure 7.3-1 (ref. 1, 2, 3).

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.

It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

The existence of "damage" is determined by radiological survey. Exceedance of the maximum ISFSI FSAR dose rates, as noted in reference 1,, 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.

The DCPP ISFSI is located wholly outside the PLANT PROTECTED AREA.

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

Background

The DCPP ISFSI Technical Specifications do not have maximum contact dose rate specified for the exterior of an overpack. The values in Table E-1 are derived from ISFSI FSAR Tables (ref 1, 2). Since the UFSAR Table 7.3-1A are the maximum calculated dose rate values, and are not expected to ever be exceeded, a conservative approach of exceeding the highest possible fuel value dose rates, plus 5 mRem/hour, was used as an indication of damage to an overpack. Note: These values are approximately 2 times the maximum expected dose rate for low burn-up fuel (ref 2).

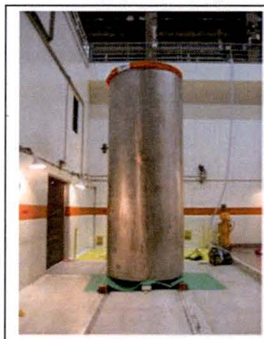
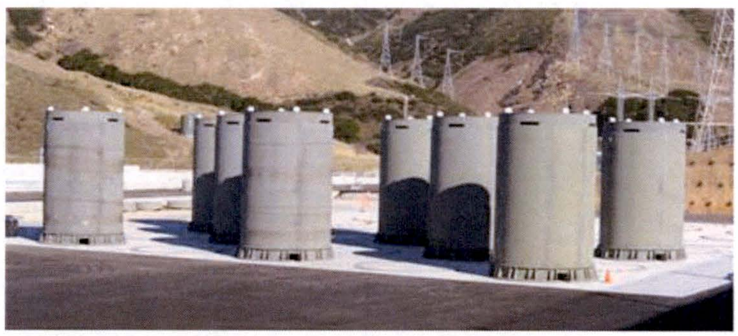
The ISFSI includes the dry-cask storage system, the cask transfer facility, onsite transporter, and the storage pads. The dry-cask storage system is the HI-STORM 100 System. This is a canister-based storage system that stores spent nuclear fuel in a vertical orientation. It consists of three discrete components: the MPC, the HI-TRAC 125 Transfer Cask, and the HI-STORM 100 System Overpack (see pictures at end of section). The MPC provides the confinement boundary for the stored fuel. The HI-TRAC 125 Transfer Cask provides radiation shielding and structural protection of the MPC during transfer operations, while the storage overpack provides radiation shielding and structural protection of the MPC during storage. The HI-STORM 100 System is passive and does not rely on any active cooling systems to remove spent fuel decay heat. After the storage casks are placed on the storage pad, the ISFSI Technical Specifications require that the casks be inspected periodically to ensure that the air vents are not blocked. Security personnel control access to the storage area and identify and assess off-normal and emergency events. Health physics personnel perform dose rate and contamination surveys to ensure that the appropriate regulatory limits are maintained. Maintenance personnel maintain the facilities including the storage casks, emergency equipment, and transport systems (ref. 4).

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.

DCCP Basis Reference(s):

1. Diablo Canyon ISFSI FSAR Update, Chapter 7 Radiation Protection, Table 7.3-1A "Surface and 1 Meter Dose Rates for the Overpack with an MPC-32 69,000 MWD/MTU and 5-Year Cooling"
2. Diablo Canyon ISFSI FSAR Update, Chapter 7 Radiation Protection, Table 7.3-1B "Surface and 1 Meter Dose Rates for the Overpack with an MPC-32 32,500 MWD/MTU and 5-Year Cooling"
3. Diablo Canyon ISFSI FSAR Update, Chapter 7, Figure 7.3-1 "Cross Section Elevation of the Generic Hi-Storm 100S Overpack with Dose Point Locations."
4. NRC Materials License No. SNM-2511, LICENSE FOR INDEPENDENT STORAGE OF SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE, Safety Evaluation Report
5. NEI 99-01 E-HU1

DCPP ISFSI HI-STORM 100 System

 MPC (CONFINEMENT BOUNDARY)	 Transporter with Hi-TRAC Transfer Cask	 HI-STORM Storage Casks (Overpack)

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 plant electrical power can compromise plant SAFETY SYSTEM operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of power to or degraded voltage on the 125V DC vital buses.

5. Loss of Communications

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

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.

DEGRADED PERFORMANCE - As applied to hazardous event thresholds, damage significant enough to cause concern regarding the operability or reliability of the affected safety system train.

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.

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 RCS inventory results in RCS water level less than a procedurally designated lower limit for ≥ 15 minutes. (Note 1)	

Note 1: The SM/SEC/ED 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):

REDUCED INVENTORY CONDITION (RIC) - The condition existing whenever RCS water level is lower than 3 feet below the reactor vessel flange (below 111-foot elevation) with fuel in the core.

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>ERO Decision Making Information</u>

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.

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.

<u>Background</u>

With the plant in Cold Shutdown, RCS water level is normally maintained above 25% Cold Calibration Pressurizer level (~129 ft. elevation). However, if RCS level is being controlled below 25%, or if level is being maintained in a procedurally 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 (ref. 2).

With the plant in Refueling mode, RCS water level is normally maintained at or above the reactor vessel flange (Technical Specifications requires at least 23 ft. of water above the top of the reactor vessel flange in the refueling cavity during refueling operations) (ref. 1). However, RCS level may be maintained below the reactor vessel flange if in "lowered inventory" or "REDUCED INVENTORY" condition (ref. 2).

This EAL addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a procedurally specified level band). This condition is considered to be a potential degradation of the level of safety of the plant.

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.

DCPP Basis Reference(s):

1. Technical Specification 3.9.7, Refueling Cavity Water Level
2. OP A-2: II, U1 Reactor Vessel - Draining the RCS to the Vessel Flange - With Fuel in Vessel
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 Structure Sumps
• Reactor Cavity Sump
• PRT
• RCDT
• CCW Surge Tank(s)
• Auxiliary Building Sump
• RWST
• RHR Room Sumps (alarm **only**)
• MEDT

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

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).

RCS LEAKAGE – RCS leakage shall be:

a. Identified Leakage

1. Leakage, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leak off), that is captured and conducted to collection systems or a sump or collecting tank;
2. 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;

3. Reactor Coolant System (RCS) leakage through a steam generator to the secondary system (primary to secondary leakage).

b. Unidentified Leakage

All leakage (except RCP seal water injection or leak off) that is not identified leakage.

c. Pressure Boundary Leakage

Leakage (except primary to secondary leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

d. 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).

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

NOTE: If an isolation valve could not be accessed due to local conditions (i.e. high radiation, temperature, etc.) then that would also make a leak unisolable, even though the inaccessible valve could isolate the leak.

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:

ERO Decision Making Information

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. The ability to monitor RCS level includes level instrumentation as well as direct and indirect (e. g. camera) visual observation.

This EAL addresses a condition where all means to determine RCS level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels (Table C-1). 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.

Background

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

In this EAL, the ability to monitor RCS level is lost such that RCS inventory loss must be detected by indirect leakage indications. The ability to monitor RCS level includes level instrumentation as well as direct and indirect (e. g. camera) visual observation. 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 to maintain RCS inventory, 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).

This IC addresses the loss of the ability to monitor RCS level concurrent with indications of RCS LEAKAGE. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

DCPP Basis Reference(s):

1. OP AP SD-2, "Loss of RCS Inventory
2. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory.

EAL:

CA1.1 Alert

Loss of RCS inventory as indicated by reactor vessel level < 107 ft. 6 in. (107.5 ft.) on RVRLIS, LI-400 standpipe or ultrasonic sensor.

OR

< 67.5% RVLIS full range (RVLIS equivalent to 107 ft. 6 in.).

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

None

Basis:

ERO Decision Making Information

When reactor vessel water level decreases to 107 ft. 6 in. el., RCS level is ~21 in. above the bottom of the RCS hot leg penetration. This is the minimum procedurally allowed RCS level to preclude vortexing of the RHR pumps while in Shutdown Cooling. This level can be monitored by:

- RVRLIS
- LI-400 standpipe
- Ultrasonic sensor

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

For this EAL, a lowering of RCS water level below the specified level 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.

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

Background

The purpose of the Reactor Vessel Refueling Level Instrumentation System (RVRLIS) is to provide reactor vessel and refueling cavity level indication during refueling, when the vessel head will be removed, and during drainage to half loop. The system is designed to be used only when the RCS is at near atmospheric pressure or when a vacuum is being established for refill operations. The wide range and narrow range RVRLIS (if required) and the LI-400 standpipe systems remain in service from the time RCS level is lowered below 25% Cold Calibrated Pressurizer level until just prior to pressurizing the RCS. Narrow Range RVRLIS is required if reduced inventory conditions (below 111 ft. elevation) are planned.

The LI-400 standpipe is a magnetic level indicator (LI-400A, B, C standpipe) and provides local indication of reactor vessel refueling level. The indicator is mounted on the outside of the secondary shield wall (crane wall) and can be viewed from the 91 ft. elevation of Containment. The indicator is composed of three mechanical flag indicator units.

RVRLIS, LI-400 standpipe and ultrasonic detectors are off-scale low (105 ft. 9 in.) when reactor vessel water level drops below the elevation of the bottom of the RCS hot leg penetration. The ultrasonic sensor is installed during an outage and measures level on one of the hot legs.

The purpose of the Reactor Vessel Level Instrumentation System (RVLIS) is to measure the level of the water or the relative void content of the coolant in the reactor vessel. The RVLIS setpoint corresponding to the minimum RHR pump operation limit was obtained as follows (ref. 2, 3, 4):

- Full range:
  - Per SC-I-87B, span = 494.9 in. and 0% = 79.6536 feet
  - % span/in. = 100 / 494.9 = 0.20206%/in. and minimum RCS level for RHR operation (from above) = 107.5 feet
  - (107.5 - 79.6536) x 12 x 0.20206 = 67.5%

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.

DCPP Basis Reference(s):

1. Plant Drawing No. 57729
2. OP A-2:III, Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in Vessel Drain-Down
3. Instrument Scaling Calculation SC-I-87B, Reactor Vessel Level Instrumentation System and Subcooled Margin Monitor
4. OP AP SD-0 Loss of, or Inadequate Decay Heat Removal
5. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory.

EAL:

CA1.2 Alert

RCS water level **cannot** be monitored for ≥ 15 minutes. (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 SM/SEC/ED should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-1 Sumps / Tanks
• Containment Structure Sumps • Reactor Cavity Sump • PRT • RCDT • CCW Surge Tank(s) • Auxiliary Building Sump • RWST • RHR Room Sumps (alarm **only**) • MEDT

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

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).

RCS LEAKAGE – RCS leakage shall be:

a. Identified Leakage

1. Leakage, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leak off), that is captured and conducted to collection systems or a sump or collecting tank;

2. 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;
3. Reactor Coolant System (RCS) leakage through a steam generator to the secondary system (primary to secondary leakage).

b. Unidentified Leakage

All leakage (except RCP seal water injection or leak off) that is not identified leakage.

c. Pressure Boundary Leakage

Leakage (except primary to secondary leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

e. 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).

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

NOTE: If an isolation valve could not be accessed due to local conditions (i.e. high radiation, temperature, etc.) then that would also make a leak unisolable, even though the inaccessible valve could isolate the leak.

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:

ERO Decision Making Information

In this EAL, the ability to monitor RCS level would be unavailable for greater than 15 minutes, and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). The ability to monitor RCS level includes level instrumentation as well as direct and indirect (e. g. camera) visual observation.

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).

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

Background

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 by direct or indirect methods, 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.

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.

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.

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

DCPP Basis Reference(s):

1. OP AP SD-2, Loss of RCS Inventory
2. OP AP-1, Excessive Reactor Coolant System Leakage
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**
With CONTAINMENT CLOSURE **not** established, RVLIS full range < 62.1%. (Note 12)	

Note 12: With RVLIS out-of-service, classification shall be based on CS1.3 or CG1.2 if RCS inventory **cannot** be monitored.

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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 Containment Closure.

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:

ERO Decision Making Information

This EAL is only applicable when RVLIS is in service (see Note 12).

When reactor vessel water level lowers to 62.1%, water level is six inches below the elevation of the bottom of the RCS hot leg penetration.

Six inches below the elevation of the bottom of the RCS hot leg penetration can be monitored only by RVLIS.

Other reactor vessel water level monitoring systems (e.g., RVRLIS, LI-400 standpipe, ultrasonic sensor, RVLIS upper range) are downscale-low when water level drops below the elevation of the bottom of the RCS hot leg penetration.

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

Background

When reactor vessel water level drops significantly below the elevation of the bottom of the RCS hot leg penetration, all sources of RCS injection have failed or are incapable of making up for the inventory loss.

The RVLIS setpoint corresponding to six inches below the elevation of the bottom of the RCS hot leg penetration was obtained as follows (ref. 1, 2, 3, 4):

- Per SC-I-87B, span = 494.9 in. and 0% = 79.6536 feet

- % span/in. = 100 / 494.9 = 0.20206%/in. and bottom of the hot leg (from above) = 105.75 feet
- (105.75 - 6 - 79.6536) x 12 x 0.20206 = 62.1%

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

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).

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.

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS/reactor vessel levels of EALs CS1.1 and CS2.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

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

DCPP Basis Reference(s):

1. Plant Drawing No. 57729
2. OP A-2:III, Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in Vessel
3. AP E-55, Equipment Elevations for RCS Flood-Up and Drain-Down
4. Instrument scaling calculation SC-I-87B, "Reactor Vessel Level Instrumentation System and Subcooled Margin Monitor
5. AD8.DC54, Containment Closure
6. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.2	**Site Area Emergency**
With CONTAINMENT CLOSURE established, RVLIS full range < 56.6% (Top of Fuel). (Note 12)	

Note 12: With RVLIS out-of-service, classification shall be based on CS1.3 or CG1.2 if RCS inventory **cannot** be monitored.

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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 Containment Closure.

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.

Bases:

ERO Decision Making Information

This EAL is only applicable when RVLIS is in service (see Note 12).

When reactor vessel water level drops below RVLIS full range indication of 56.6% core uncovery is about to occur. This level drop can only be remotely monitored by reactor vessel Level Instrumentation System (RVLIS).

Other reactor vessel water level monitoring systems (e.g., RVRLIS, LI-400 standpipe, ultrasonic sensor, RVLIS upper range) are downscale-low when water level drops below the elevation of the bottom of the RCS hot leg penetration.

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

Background

The RVLIS setpoint corresponding to the top of fuel was obtained as follows (ref. 1, 2, 3, 4):

- Per SC-I-87B, span = 494.9 in. and 0% = 79.6536 feet
- % span/in. = 100 / 494.9 = 0.20206%/in. and top of core = 103 feet
- (103 - 79.6536) x 12 x 0.20206 = 56.6%

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

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).

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.

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

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.

DCPP Basis Reference(s):

1. Plant Drawing No. 57729
2. OP A-2:III, Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in Vessel
3. AP E-55, Equipment Elevations for RCS Flood-Up and Drain-Down
4. Instrument scaling calculation SC-I-87B, "Reactor Vessel Level Instrumentation System and Subcooled Margin Monitor
5. AD8.DC54, Containment Closure
6. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.3 Site Area Emergency

RCS water level **cannot** be monitored for ≥ 30 minutes. (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 uncover.
- **Any** Bridge (Manipulator) Crane Radiation Monitor > 9 R/hr.
- Erratic Source Range Monitor indication.

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

Table C-1 Sumps / Tanks
• Containment Structure Sumps
• Reactor Cavity Sump
• PRT
• RCDT
• CCW Surge Tank(s)
• Auxiliary Building Sump
• RWST
• RHR Room Sumps (alarm **only**)
• MEDT

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

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

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).

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:

ERO Decision Making Information

In this EAL, the ability to monitor RCS level would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). The ability to monitor RCS level includes level instrumentation as well as direct and indirect (e. g. camera) visual observation.

The reactor vessel inventory loss may be detected by the radiation monitors or erratic source range monitor indication. As water level in the reactor vessel lowers, the dose rate above the core will rise. The Bridge Crane Radiation Monitors can be monitored either locally, or remotely on the Viewpoint software.

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.1).

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 .

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

Background

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.

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.

The dose rate due to this core shine should result in increased Bridge (Manipulator) Crane Radiation Monitor indication. A reading of 9 R/hr (90% of instrument scale) is indicative of core uncovery. There are a number of variables governing the projected dose rate from an actual core uncover (ref. 2).

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.

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.

DCPP Basis Reference(s):

1. Nuclear Safety Analysis Center (NSAC), 1980, "Analysis of Three Mile Island - Unit 2 Accident," NSAC-1
2. EP-EALCALC-DCPP-1603 Radiation Monitor Readings for Core Uncovery During Refueling
3. 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

RVLIS full range < 56.6% (Top of Fuel) for ≥ 30 minutes. (Notes 1, 12)

AND

Any Containment Challenge indication, Table C-2.

Note 1: The SM/SEC/ED 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.

Note 12: With RVLIS out-of-service, classification shall be based on CS1.3 or CG1.2 if RCS inventory **cannot** be monitored.

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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 Containment Closure.

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>ERO Decision Making Information</u>

This EAL is only applicable when RVLIS is in service (see Note 12).

When reactor vessel water level drops below RVLIS full range indication of 56.6% core uncovery is about to occur. This level drop can only be remotely monitored by Reactor Vessel Level Instrumentation System (RVLIS).

Other reactor vessel water level monitoring systems (e.g., RVRLIS, LI-400 standpipe, ultrasonic sensor, RVLIS upper range) are downscale-low when water level drops below the elevation of the bottom of the RCS hot leg penetration.

Three conditions are associated with a challenge to Containment:

1. CONTAINMENT COSURE not established
2. Containment hydrogen ≥ 4%
3. UNPLANNED rise in Containment pressure (Containment pressure changes due to ventilation system changes do not constitute a containment challenge)

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.

During periods when installed containment hydrogen gas monitors are out-of-service, 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.

Background

The RVLIS setpoint corresponding to the top of fuel was obtained as follows (ref. 1, 2, 3, 4):

- Per SC-I-87B, span = 494.9 in. and 0% = 79.6536 feet
- % span/in. = 100 / 494.9 = 0.20206%/in. and top of core = 103 feet
- (103 - 79.6536) x 12 x 0.20206 = 56.6%

Three conditions are associated with a challenge to Containment:

1. CONTAINMENT COSURE 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 be required.
2. Containment hydrogen ≥ 4% - The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations. To generate such levels of

combustible gas, loss of the Fuel Clad and RCS barriers are likely to have occurred. Operation of the Containment Hydrogen Recombiner with Containment hydrogen concentrations above 4.0% could result in ignition of the hydrogen. If in operation, containment hydrogen can be monitored in the Control Room on ANR-82/ANR-83 and PAM1 following local equipment initialization (ref. 6, 7)

3. UNPLANNED rise in Containment pressure - In the operating modes associated with this EAL, Containment pressure is expected to remain very low; thus, an elevated Containment pressure resulting from an UNPLANNED rise above near-atmospheric pressure conditions may be indicative of a challenge to the Containment barrier. Containment pressure changes due to ventilation system changes do not constitute a containment challenge.

Under the conditions specified by this EAL, continued lowering of reactor vessel water level is indicative of a loss of inventory control with a challenge to the Containment. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the reactor vessel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS or reactor vessel water level drop and potential core uncovery. The inability to restore and maintain level inventory within 30 minutes after reaching this condition in combination with a Containment challenge infers a failure of the RCS barrier, Loss of the Fuel Clad barrier and a Potential Loss of Containment.

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. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

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

The 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.

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*.

DCPP Basis Reference(s):

1. Plant Drawing No. 57729

2. OP A-2:III, Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in Vessel
3. AP E-55, Equipment Elevations for RCS Flood-Up and Drain-Down
4. Instrument scaling calculation SC-I-87B, "Reactor Vessel Level Instrumentation System and Subcooled Margin Monitor
5. AD8.DC54, Containment Closure
6. CA-3, Hydrogen Flammability in Containment
7. OP H-9, INSIDE CONT H2 RECOMB SYSTEM
8. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CG1.2 General Emergency

RCS level **cannot** be monitored for ≥ 30 minutes. (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 uncover.
- **Any** Bridge (Manipulator) Crane Radiation Monitor > 9 R/hr.
- Erratic Source Range Monitor indication.

AND

Any Containment Challenge indication, Table C-2.

Note 1: The SM/SEC/ED 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 Structure Sumps • Reactor Cavity Sump • PRT • RCDT • CCW Surge Tank(s) • Auxiliary Building Sump • RWST • RHR Room Sumps (alarm **only**) • MEDT

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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 Containment Closure.

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).

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:

ERO Decision Making Information

In this EAL, the ability to monitor RCS level would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). The ability to monitor RCS level includes level instrumentation as well as direct and indirect (e. g. camera) visual observation.

The reactor vessel inventory loss may be detected by the radiation monitors or erratic source range monitor indication. As water level in the reactor vessel lowers, the dose rate above the core will rise. The Bridge Crane Radiation Monitors can be monitored either locally, or remotely on the Viewpoint software.

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. 1).

Source Range indication can be seen on Source Range Detectors NI-31 & 32 as well as the Gammametrics detectors.

Three conditions are associated with a challenge to Containment:

1. CONTAINMENT COSURE not established

2. Containment hydrogen ≥ 4%
3. UNPLANNED rise in Containment pressure (Containment pressure changes due to ventilation system changes do not constitute a containment challenge)

During periods when installed containment hydrogen gas monitors are out-of-service, 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.

Background

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.

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.

The dose rate due to this core shine should result in increased Bridge (Manipulator) Crane Radiation Monitor indication. A reading of 9 R/hr (90% of instrument scale) is indicative of core uncovery. There are a number of variables governing the projected dose rate from an actual core uncover (ref. 5).

Three conditions are associated with a challenge to Containment:

1. CONTAINMENT COSURE 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.2). If containment closure is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to GE would not be required.
2. Containment hydrogen ≥ 4% - The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations. To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers are likely to have occurred. Operation of the Containment Hydrogen Recombiner with Containment hydrogen concentrations above 4.0% could result in ignition of the hydrogen. Containment hydrogen can be monitored in the Control Room on ANR-82/ANR-83 and PAM1 following local equipment initialization (ref. 3, 4)

3. UNPLANNED rise in Containment pressure - In the operating modes associated with this EAL, Containment pressure is expected to remain very low; thus, an elevated Containment pressure resulting from an UNPLANNED rise above near-atmospheric pressure conditions may be indicative of a challenge to the Containment barrier. Containment pressure changes due to ventilation system changes do not constitute a containment challenge.

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.

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*.

DCPP Basis Reference(s):

1. Nuclear Safety Analysis Center (NSAC), 1980, "Analysis of Three Mile Island - Unit 2 Accident," NSAC-1
2. AD8.DC54, Containment Closure
3. OP H-9, INSIDE CONT H2 RECOMB SYSTEM
4. CA-3, Hydrogen Flammability in Containment
5. EP-EALCALC-DCPP-1603 Radiation Monitor Readings for Core Uncovery During Refueling
6. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Vital AC Power

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

EAL:

CU2.1 Unusual Event

AC power capability, Table C-3, to Unit 1 or Unit 2 vital 4.16KV buses F, G and H reduced to a single power source for ≥ 15 minutes. (Note 1)

AND

A failure of that single power source will result in loss of **all** AC power to SAFETY SYSTEMS.

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

Table C-3 AC Power Capability

	Unit 1	Unit 2
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

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:

ERO Decision Making Information

For emergency classification purposes, “capability” means that, whether or not the buses are actually powered from it, an AC power source(s) can be aligned to the vital buses within 15 minutes:

- By a clear procedure path,
  and
- Breakers and equipment are readily available to power up the bus within the allotted time frame.

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.

An “AC power source” is a source recognized in AOPs and EOPs, and capable of supplying required power to a vital 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 vital 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 vital buses being back-fed from an offsite power source.
- If the 1-1- or 2-1 transformer is unavailable, the other unit's #1 transformer (2-1 or 1-1) can be used to supply the startup bus through the startup bus cross tie breaker.

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

When filling out the ENF form, this event can be Unit 1, Unit 2 or Unit 1 and 2.

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

Background

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 vital buses.

4.16KV buses F, G and H are the emergency (vital) buses. Each bus has three sources of power (see figure below).

One offsite source is from the 230KV switchyard via 12KV startup transformer 1-1 (2-1) to the 4.16KV startup transformer 1-2 (2-2).

Another method to obtain offsite power is by back feeding the vital buses from the 500KV switchyard through the main transformer to the 4.16KV unit auxiliary transformer 1-2 (2-2). This is normally only done post-trip when 500KV power is available.

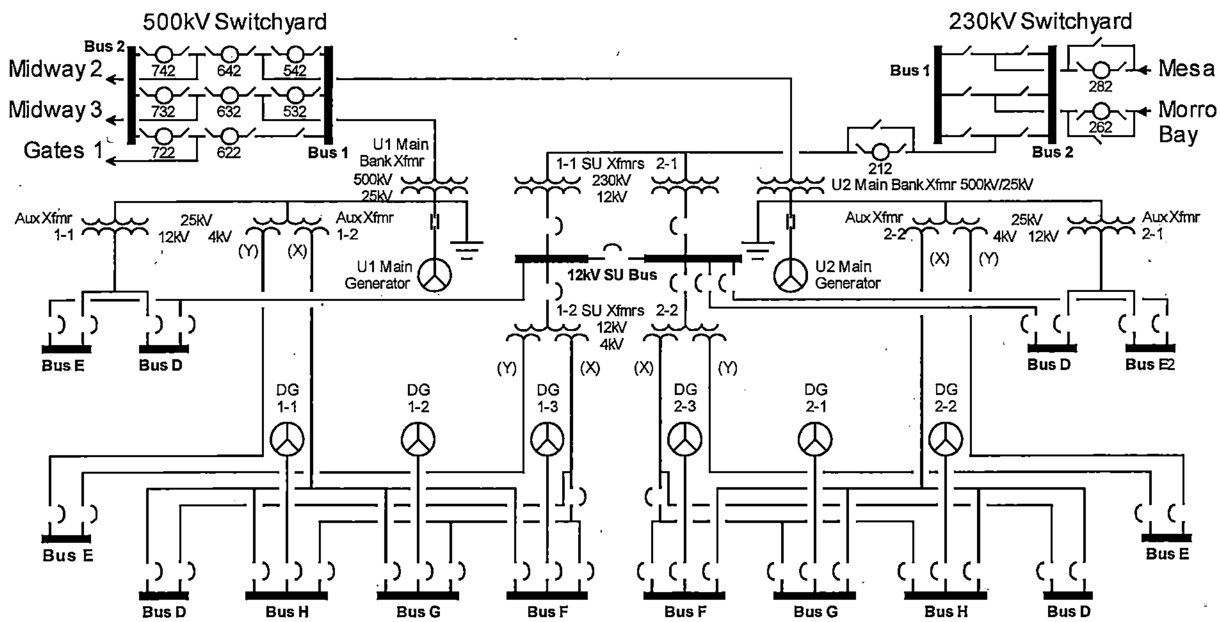
In addition, each units vital buses F, G and H have an onsite emergency diesel generator which can supply electrical power to its associated bus automatically in the event that the preferred source becomes unavailable (ref. 1-6).

Refer to CP M-10 Fire Protection of Safe Shutdown Equipment for a list of SAFETY SYSTEMS.

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.

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

DCPP Electrical Distribution System



DCPP Basis Reference(s):

1. UFSAR, Section 8.2.2
2. UFSAR, Section 8.3.1.6
3. OP AP SD-1, Loss of AC Power
4. OP AP-2, Loss of Offsite Power
5. OP J-2:V, Backfeeding the Unit From the 500kV System
6. ECA-0.0, Loss of All Vital AC Power
7. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Vital AC Power

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

EAL:

CA2.1	**Alert**
Loss of **all** offsite and **all** onsite AC power capability, Table C-3, to Unit 1 or Unit 2 vital 4.16KV buses F, G and H for ≥ 15 minutes. (Note 1)	

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

Table C-3 AC Power Capability

	Unit 1	**Unit 2**
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

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:

<u>ERO Decision Making Information</u>

For emergency classification purposes, "capability" means that, whether or not the buses are actually powered from it, an AC power source(s) can be aligned to the vital buses within 15 minutes:

- By a clear procedure path,
  and
- Breakers and equipment are readily available to power up the bus within the allotted time frame.

This EAL addresses a total loss of AC power for greater than 15 minutes 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.

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

When filling out the ENF form, this event can be Unit 1, Unit 2 or Unit 1 and 2.

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

<u>Background</u>

The vital 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant (see figure below).

Unit 1(2) 4.16KV buses F, G and H are the emergency (vital) buses. Each bus has three sources of power.

One offsite source is from the 230KV switchyard via 12KV startup transformer 1-1 (2-1) to the 4.16KV startup transformer 1-2 (2-2).

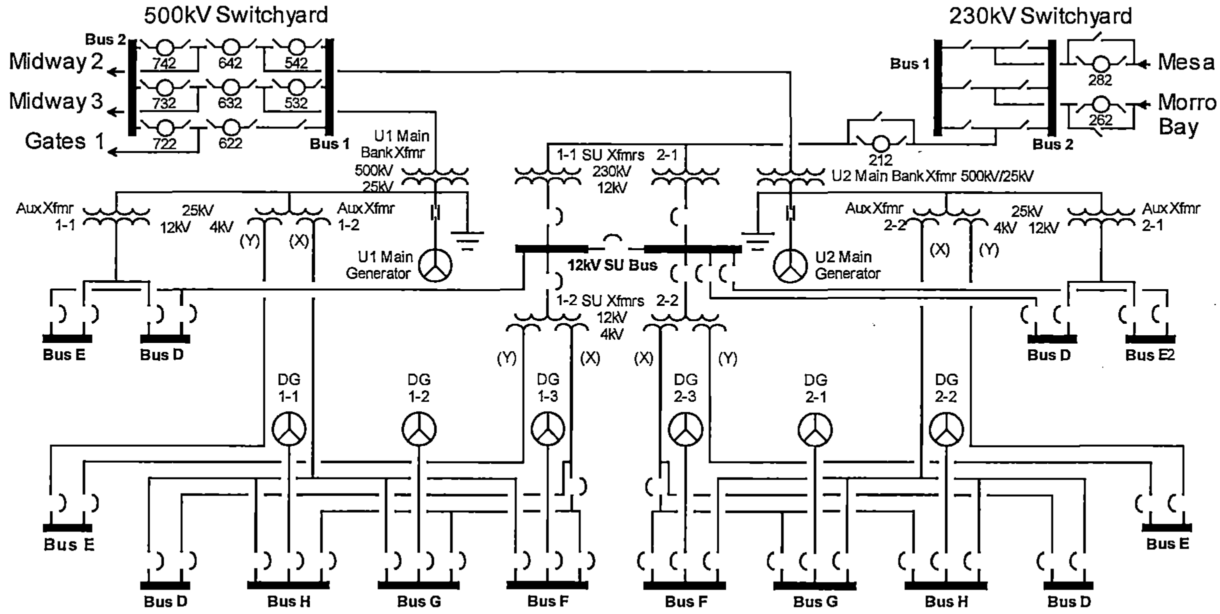
Another method to obtain offsite power is by back feeding the vital buses from the 500KV switchyard through the main transformer to the 4.16KV unit auxiliary transformer 1-2 (2-2). This is normally only done post-trip when 500KV power is available.

In addition, each units vital buses F, G and H have an onsite emergency diesel generator which can supply electrical power to its associated bus automatically in the event that the preferred source becomes unavailable (ref. 1-6).

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 a vital 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.

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

DCPP Electrical Distribution System



DCPP Basis Reference(s):

1. UFSAR, Section 8.2.2
2. UFSAR, Section 8.3.1.6
3. OP AP SD 1, Loss of AC Power
4. OP AP-2, Loss of Offsite Power
5. OP J-2:V, Backfeeding the Unit From the 500kV System
6. ECA-0.0, Loss of All Vital AC Power
7. 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.

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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 "Containment Closure."

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).

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:

ERO Decision Making Information

In the absence of reliable RCS temperature indication caused by a loss of decay heat removal capability, classification should be based on EAL CU3.2 should RCS level indication be subsequently lost.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit of 200°F when the heat removal function is available does not warrant a classification.

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

Background

Numerous instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1). These may include but are not limited to (ref. 2):

- TR413 Loop 1 Wide Range Temperature
- TR423 Loop 2 Wide Range Temperature

- TR433 Loop 3 Wide Range Temperature
- TR443 Loop 4 Wide Range Temperature
- WR T_{hot} recorders 0-700°F
- WR T_{cold} recorders 0-700°F
- CETs
- RHR System temperatures (when RHR is in service)

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 SM/SEC/ED should also refer to IC CA3.

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.

DCPP Basis Reference(s):

1. DCPP Technical Specifications Table 1.1-1 Modes
2. OP L-1, Plant Heatup From Cold Shutdown to Hot Standby
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 **all** RCS level indication for ≥ 15 minutes. (Note 1)	

Note 1: The SM/SEC/ED 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):

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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 "Containment Closure."

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).

Basis:

ERO Decision Making Information

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 SM/SEC/ED should also refer to IC CA3.

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.

Background

Reactor vessel water level is normally monitored using the following instruments (ref. 1):

- RVRLIS
- LI-400 Standpipe
- RVLIS
- Ultrasonic level detectors

AP E-55, "Equipment Elevations for RCS Flood-Up and Drain-Down", provides a cross-reference of indicated water levels and key plant elevations

Numerous instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 2). These may include but are not limited to (ref. 3):

- TR413 Loop 1 Wide Range Temperature
- TR423 Loop 2 Wide Range Temperature
- TR433 Loop 3 Wide Range Temperature
- TR443 Loop 4 Wide Range Temperature
- WR T_{hot} recorders 0-700°F
- WR T_{cold} recorders 0-700°F
- CETs
- RHR System temperatures (when RHR is in service)

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.

DCPP Basis Reference(s):

1. AP E-55, "Equipment Elevations for RCS Flood-Up and Drain-Down
2. DCPP Technical Specifications Table 1.1-1
3. OP L-1, Plant Heatup From Cold Shutdown to Hot Standby
4. 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 does **not** apply during water-solid plant conditions).

Note 1: The SM/SEC/ED 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.

Table C-4: RCS Heat-up Duration Thresholds

RCS Status	CONTAINMENT CLOSURE Status	Heat-up Duration
INTACT (but **not** REDUCED INVENTORY)	N/A	60 minutes*
Not INTACT **OR** REDUCED INVENTORY	established	20 minutes*
	not established	0 minutes

* If an RCS heat removal system is in operation within this time frame and RCS temperature is trending down, 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 DCPP, Containment Closure is defined by Administrative Procedure AD8.DC54 Containment Closure.

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).

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.

REDUCED INVENTORY - The condition existing whenever RCS water level is lower than 3 feet below the reactor vessel flange (below 111-foot elevation) with fuel in the core.

Basis:

ERO Decision Making Information

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 the RCS is INTACT in Mode 5 or based on time to boil data when in Mode 6 or the RCS is not INTACT in Mode 5.

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.

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

Background

Numerous instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1). These may include but are not limited to (ref. 2):

- TR413 Loop 1 Wide Range Temperature
- TR423 Loop 2 Wide Range Temperature
- TR433 Loop 3 Wide Range Temperature
- TR443 Loop 4 Wide Range Temperature
- WR T_{hot} recorders 0-700°F
- WR T_{cold} recorders 0-700°F
- CETs
- RHR System temperatures (when RHR is in service)

PI-403A, PI-405 and PI-405A display on VB2, with digital values available on PPC, SPDS and SCMM. Digital readouts can display changes of less than 10 psig.

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.

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 in the absence of RCS temperature monitoring capability.

DCPP Basis Reference(s):

1. DCPP Technical Specifications Table 1.1-1
2. OP L-1, Plant Heatup From Cold Shutdown to Hot Standby
3. 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**
< 105 VDC bus voltage indications on Technical Specification **required** 125 VDC vital buses for ≥ 15 minutes. (Note 1)	

Note 1: The SM/SEC/ED 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):

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:

ERO Decision Making Information

Minimum battery voltage of 105 VDC is the voltage below which supplied loads may not function (ref. 2, 3, 4).

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.

Background

The vital 125 VDC system consists of three independent networks per unit. Each network contains the following components:

- Battery
- Battery charger
- Standby battery charger to allow maintenance and/or testing
- Distribution panels
- Ground detector

The vital 125 VDC batteries and distribution panels are located in Area "H" of the 115 feet elevation of the Auxiliary Building. There are a total of three batteries per unit, 11(21), 12(22), and 13(23). The batteries are sized to provide sufficient power to operate the associated DC loads for the time necessary to safely shut down the unit, should a 480-VAC source to one or more battery chargers be unavailable (ref. 1, 2, 3).

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.

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

DCPP Basis Reference(s):

1. UFSAR, Section 8.3.2.2.2
2. OP AP-23, Loss of Vital DC Bus
3. ECA-0.0, Loss of All Vital AC Power
4. Notification 50804190 DC Bus Voltage Trigger for EALs
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 offsite communication methods.

OR

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

Table C-5 Communication Methods

System	Onsite	Offsite	NRC
Unit 1, Unit 2 and TSC Radio Consoles	X	X	
DCPP Telephone System (PBX)	X	X	X
Portable radio equipment (handie-talkies)	X		
Operations Radio System	X	X	
Security Radio Systems	X		
CAS and SAS Consoles	X	X	X
Fire Radio System	X		
Hot Shutdown Panel Radio Consoles	X	X	X
Public Address System	X		
NRC FTS			X
Mobile radios	X		
Satellite phones	X	X	X
Direct line (ATL) to the County and State OES		X	

Mode Applicability:

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

Definition(s):

None

Basis:

ERO Decision Making Information

Onsite, offsite and NRC communications include one or more of the systems listed in Table C-5 (ref. 1, 2, 3).

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

This IC addresses a significant loss of onsite or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to Offsite Response Organizations (OROs) and the NRC.

Background

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 county EOCs

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

DCPP Basis Reference(s):

1. UFSAR, Section 9.5.2
2. Emergency Plan Section 7.2 Communications Equipment
3. AR PK15-23, Communications
4. 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 • Tsunami • Other events with similar hazard characteristics as determined by the SM/SEC/ED

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

DEGRADED PERFORMANCE – As applied to hazardous event thresholds, event damage significant enough to cause concern regarding the operability or reliability of the affected safety system train.

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.

TORNADO - A violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

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:

<u>ERO Decision Making Information</u>

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.

The indications of DEGRADED PERFORMANCE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

In modes 5, 6 and defueled, the appropriate plant configuration based Outage Safety Checklist in AD8.DC55 "Outage Safety Scheduling" should be consulted to identify required equipment supporting each of the specified safety functions (ref. 1).

With respect to event damage caused by an equipment failure resulting in a FIRE or EXPLOSION, no emergency classification is required in response to a FIRE or EXPLOSION resulting from an equipment failure if the only safety system equipment affected by the event is that upon which the failure occurred. An emergency classification is required if a FIRE or EXPLOSION caused by an equipment failure damages safety system equipment that was otherwise functional or operable (i.e., equipment that was not the source/location of the failure). For example, if a FIRE or EXPLOSION resulting from the failure of a piece of safety system equipment causes damage to the other train of the affected safety system or another safety system, then an emergency declaration is required in accordance with this IC and EAL.

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

Background

This condition represents an actual or potential substantial degradation of the level of safety of the plant. Due to this actual or potential substantial degradation, this condition can significantly reduce the margin to a loss or potential loss of a fission product barrier.

The first conditional addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available.

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.

DCPP Basis Reference(s):

1. AD8.DC55 Outage Safety Scheduling
2. 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 PLANT PROTECTED AREA, bomb threats, sabotage attempts, and actual security compromises threatening loss of physical control of the plant.

2. Seismic Event

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

3. Natural or 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 ISFSI or PLANT PROTECTED AREA or which may affect operability of equipment needed for safe shutdown

5. Control Room Evacuation

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

6. SM/SEC/ED Judgment

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

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):

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

OWNER CONTROLLED AREA (OCA) - For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan.

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.

SECURITY EVENT - Any incident representing an attempted, threatened, or actual breach of the security system or reduction of the operational effectiveness of that system. A security event can result in either a SECURITY CONDITION or HOSTILE ACTION.

Basis:

ERO Decision Making Information

The intent of the EAL is to ensure that appropriate notifications for the security threat are made in a timely manner. The DCPP Security and Safeguards Contingency Plan provides a description of SECURITY EVENTS indicative of a potential loss of the level of safety of the plant. Events at the Unusual Event level include credible threats to attack or use a bomb against the plant, or involve extortion, coercion or HOSTAGE threats.

NOTE: **DO NOT** revise this Technical Basis Document to add any identifying information to any SECURITY EVENT codes, and do not remove this note.

Threats under this EAL include the following SECURITY EVENT categories:

- SE-1, SE-2, SE-3, SE-7, SE-9, SE-10, SE-11, SE-12, SE-13, SE-14, SE-15, SE-16, SE-17, SE-18, SE-19, SE-20 & SE-21

Security Watch Commanders are the designated on-site 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 DCPP Security and Safeguards Contingency Plan (Safeguards) information. (ref. 1).

The first threshold:

The Security Watch Commanders, as the trained individuals confirm that a SECURITY EVENT is occurring or has occurred, and whether or not the event **is** or **is not** a HOSTILE ACTION. Training on SECURITY EVENT confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.39 information.

The second threshold:

The receipt of a credible security threat is assessed in accordance with the Security and Safeguards Contingency Plan (ref. 1). This EAL is met when the plant receives information from the NRC or other reliable source, such as the FBI.

The third threshold:

This EAL is met when the plant receives information regarding an aircraft threat from the NRC or other reliable source, such as the FBI, FAA, or NORAD, and the aircraft is more than 30 minutes away from the plant. In this EAL the threat from the impact of an aircraft on the plant is assessed. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with the Security and Safeguards Contingency Plan.

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

Background

The security shift supervision is defined as the Security Watch Commander.

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.

Threat information may come from various sources, including the NRC or FBI. Only the plant to which the specific threat is made need declare the Unusual Event.

This EAL is based on the DCPP Security and Safeguards Contingency 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, as outlined in DCPP Administrative Procedure XI1.ID2 (ref. 3).

SECURITY EVENTS assessed as HOSTILE ACTIONS are classifiable under ICs HA1 and HS1.

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*.

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 and Safeguards Contingency Plan.

DCPP Basis Reference(s):

1. DCPP Security and Safeguards Contingency Plan
2. DCPP Procedures (Procedure names and designations are controlled due to the nature of Safeguards and 10 CFR § 2.39 information.)
3. DCPP Administrative Procedure XI1.ID2 "Regulatory Reporting Requirements and Reporting Process"
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

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

Mode Applicability:

All

Definition(s):

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

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

OWNER CONTROLLED AREA (OCA) - For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan.

PLANT PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

<u>ERO Decision Making Information</u>

The intent of the EAL is to ensure that appropriate notifications are made in a timely manner. The DCPP Safeguards Contingency Plan provides a description of SECURITY EVENTS indicative of a potential loss of the level of safety of the plant.

NOTE: DO NOT revise this Technical Basis Document to add any identifying information to any SECURITY EVENT codes, and do not remove this note.

Threats under this EAL include the following SECURITY EVENT categories:

- SE-1, SE-2, SE-5, SE-10, SE-16, SE-18 & SE-19

Security Watch Commanders are the designated on-site 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 DCPP Security and Safeguards Contingency Plan (Safeguards) information. (ref. 1).

The first threshold:

Is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA (OCA). This includes any action directed against an ISFSI that is located outside the PLANT PROTECTED AREA.

This event will require rapid response and assistance due to the possibility of the attack progressing to the PLANT PROTECTED AREA.

The OCA is the area and boundary contained in the DCPP Security and Safeguards Contingency Plan (ref. 1). Generally described, it is the area between Security Gate A (aka North Gate, and is located on the road located at the north edge of the exclusion area/SITE BOUNDARY) to Security Gate E (located on the main access road just north of Secondary (Backup) Met Tower and the SITE BOUNDARY), and extending eastward to encompass the 500 and 230kV switchyards, and bounded on the west by the Pacific Ocean. On UFSAR Figure 2.1-2 this is approximated as the "Exclusion Area Boundary". A copy of UFSAR Figure 2.1-2 is at the end of definitions section of this document.

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 as outlined in DCPP Administrative Procedure XI1.ID2 (ref. 3).

The second threshold:

An assessment of 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.

This event will require rapid response and assistance due to the possibility of the need to prepare the plant and staff for a potential aircraft impact.

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.

Background

The security shift supervision is defined as the Security Watch Commander (ref. 1).

Timely and accurate communications between the Security Watch Commander 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.

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 DCPP Security and Safeguards Contingency Plan.

DCPP Basis Reference(s):

1. DCPP Security and Safeguards Contingency Plan
2. DCPP Procedures (Procedure names and designations are controlled due to the nature of Safeguards and 10 CFR § 2.39 information.)
3. DCPP Administrative Procedure XI1.ID2 “Regulatory Reporting Requirements and Reporting Process”
4. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the PLANT PROTECTED AREA.

EAL:

HS1.1	**Site Area Emergency**
A HOSTILE ACTION is occurring or has occurred within the PLANT PROTECTED AREA as reported by the Security Watch Commander.	

Mode Applicability:

All

Definition(s):

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

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

OWNER CONTROLLED AREA (OCA) - For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan.

PLANT PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

Basis:

<u>ERO Decision Making Information</u>

The intent of this EAL is to address the potential for a very rapid progression of events due to a dedicated attack. It is not intended to address incidents that are accidental or acts of civil disobedience, such as physical disputes between employees within the OCA or PLANT PROTECTED AREA. Those events are adequately addressed by other EALs. HOSTILE ACTION identified above encompasses various acts including SECURITY EVENTS:

NOTE: DO NOT revise this Technical Basis Document to add any identifying information to any SECURITY EVENT codes, and do not remove this note.

Threats under this EAL include the following SECURITY EVENT categories:

- SE-2, SE-4, SE-5, SE-10, SE-16

This class of SECURITY EVENTS represents an escalated threat to plant safety above that contained in the Alert IC in that a hostile force has progressed from the OWNER CONTROLLED AREA (OCA) to the PLANT PROTECTED AREA (PA). Although DCPP security officers are well trained and prepared to protect against hostile action, it is appropriate for Offsite Response Organizations (OROs) to be notified and encouraged to begin preparations for public protective actions (if they do not normally) to be better prepared should it be necessary to consider further actions.

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

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 as outlined in DCPP Administrative Procedure XI1.ID2 (ref. 3).

Security Watch Commanders are the designated on-site 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 DCPP Security and Safeguards Contingency Plan (Safeguards) information. (ref. 1).

<u>Background</u>

The security shift supervision is defined as the Security Watch Commander.

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

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

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

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 DCPP Security and Safeguards Contingency Plan.

DCPP Basis Reference(s):

1. DCPP Security and Safeguards Contingency Plan

2. DCPP Procedures (Procedure names and designations are controlled due to the nature of Safeguards and 10 CFR § 2.39 information.)
3. DCPP Administrative Procedure XI1.ID2 "Regulatory Reporting Requirements and Reporting Process"
4. NEI 99-01 HS1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than Design Earthquake (DE) level.

EAL:

HU2.1	**Unusual Event**
Seismic event > DE PGA as indicated by ground acceleration > 0.2 g on the "X" or "Y" axis or > 0.133 g on the "Z" axis. (Note 11)	

Note 11: If the Earthquake Force Monitor (EFM) is out of service, refer to CP M-4 Earthquake for alternative methods to assess earthquakes.

Mode Applicability:

All

Definition(s):

None

Basis:

ERO Decision Making Information

Ground motion acceleration > 0.2 g on the "X" or "Y" axis or > 0.133g on the "Z" axis is the peak ground acceleration (PGA) criterion for a Design Earthquake (DE) (ref. 3).

If the EFM indicator alarms (> 0.2 g on the "X" or "Y" axis or > 0.133g on the "Z" axis) indicating the DE PGA has been exceeded, an Unusual Event should be declared. The "X" and "Y" axes correspond to horizontal peak acceleration values while the "Z" axis corresponds to vertical peak acceleration values.

If the EFM is not operable, the earthquake magnitude is determined by alternative methods in accordance with CP M-4, "Earthquake." If it is determined that any peak acceleration has exceeded 0.2 g on the "X" or "Y" axis or 0.133g on the "Z" axis, an Unusual Event should be declared (ref. 3).

Event verification with external sources should not be necessary during or following a DE. 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 SM/SEC/ED may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

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

Background

In the event of an earthquake measuring greater than or equal to 0.01 g, the Seismic Instrumentation System annunciator PK15-24 will alert the control room and peak acceleration indications will be displayed on the EFM. The primary means for timely determination of the magnitude of an earthquake, and subsequently assessing emergency action levels, is using the EFM located in the control room (ref. 2).

When the seismic monitoring system alarms, SM directs actions as defined in CP M-4, "Earthquake," and the seismic instrumentation system engineer is notified to coordinate post-earthquake activities including retrieval and analysis of the seismic event data. The purpose of the analysis is to determine within 4 hours whether the computed response spectra associated with any of the three directional components of the seismic event exceed the DE response spectra exceedance criterion (ref.4).

It should be noted that the DE PGA values are the zero period accelerations associated the DE response spectra. Since the DE PGA indications are available and displayed on the EFM within minutes, these are the indications used for timely emergency classification. The seismic monitoring system also stores the seismic event data and generates reports later used during the post-earthquake evaluation (ref.4)

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. Such confirmation should not, however, preclude a timely emergency declaration based on receipt of the EFM alert alarm. 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 DCPP. Alternatively, near real-time seismic activity can be accessed via the NEIC website:

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

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for a Design Earthquake (DE). An earthquake greater than a DE but less than a Double Design Earthquake (DDE) 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.

NOTE: An Operating Basis Earthquake (OBE) is referred to as Design Earthquake (DE) at DCPP, and a Safe Shutdown Earthquake (SSE) is referred to as Double Design Earthquake (DDE) at DCPP (ref. 3).

DCPP Basis Reference(s):

1. DCM T-6, Seismic Analysis of Structures
2. AR PK 15-24, Seismic Instr System
3. CP M-4, Earthquake
4. AWP E-017 Guidelines for Post-Earthquake Engineering Response
5. 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 PLANT PROTECTED AREA.	

Mode Applicability:

All

Definition(s):

PLANT PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

TORNADO - A violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

Basis:

ERO Decision Making Information

A TORNADO striking (touching down) within the PLANT PROTECTED AREA warrants declaration of an Unusual Event regardless of the measured wind speed at the meteorological tower.

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.

Background

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 PLANT PROTECTED AREA.

DCPP Basis Reference(s):

1. CP M-16 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 required for the current operating mode. (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.

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:

ERO Decision Making Information

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.

In modes 5, 6 and defueled, the appropriate plant configuration based Outage Safety Checklist in AD8.DC55 “Outage Safety Scheduling” should be consulted to identify required equipment supporting each of the specified safety functions (ref. 1).

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

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

Background

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

DCPP Basis Reference(s):

1. AD8.DC55 Outage Safety Scheduling
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:

HU3.3	**Unusual Event**

Movement of personnel within the PLANT PROTECTED AREA is IMPEDED due to an event involving hazardous materials (e.g., a chemical spill or toxic gas release from an area outside the PLANT PROTECTED AREA).

Mode Applicability:

All

Definition(s):plant

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

PLANT PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

Basis:

ERO Decision Making Information

This EAL is applicable to events in areas external to the DCPP PLANT PROTECTED AREA.

This EAL addresses a hazardous materials event originating outside the PLANT PROTECTED AREA and of sufficient magnitude to IMPEDE the movement of personnel within the PLANT PROTECTED AREA.

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

Background

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

DCPP Basis Reference(s):

1. CP M-9A Hazardous Material Incident – Initial Emergency Response/Mitigation Procedure
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:

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

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

Mode Applicability:

All

Definition(s):

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

Basis:

ERO Decision Making Information

This EAL addresses a hazardous event that causes an impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include when both north and south access routes are unavailable due to site FLOODING caused by a hurricane, heavy rains, dam failure, tsunami, mudslide, etc., blocking the access and egress roads (refer to CP M-12).

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

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

Background

Refer to CP M-12 Stranded Plant for conditions in which viable plant access routes are lost (ref. 1).

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

DCPP Basis Reference(s):

1. CP M-12 Stranded Plant
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.1 Unusual Event

A FIRE is **not** extinguished within 15 minutes 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 SM/SEC/ED should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Containment • Auxiliary Building • Fuel Handling Building • Turbine Building • Intake Structure Lower Levels • Pipe Rack • Main, Auxiliary & Startup Transformers

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.

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:

ERO Decision Making Information

Multiple flow switches for the same general vicinity constitute a single alarm. This is because water flow in the sprinkler system can be seen on multiple switches for the same location. However, smoke and flame detectors are all individual alarms spaced far enough apart that each should be considered independent of each other.

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 HU4.1 assessment purposes, the emergency declaration clock starts at the time that multiple alarms or indications are received, the report was received, or the time that a single alarm is confirmed by subsequent verification action. 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.

Background

Table H-1 Fire Areas are based on CP M-10, Fire Protection of Safe Shutdown Equipment. 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.

DCPP Basis Reference(s):

1. CP M-10, Fire Protection of Safe Shutdown Equipment
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 associated with **any** Table H-1 area.

AND

The existence of a FIRE is **not** verified within 30 minutes of alarm receipt. (Note 1)

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

Table H-1 Fire Areas
• Containment • Auxiliary Building • Fuel Handling Building • Turbine Building • Intake Structure Lower Levels • Pipe Rack • Main, Auxiliary & Startup Transformers

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.

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:

ERO Decision Making Information

Multiple flow switches for the same general vicinity constitute a single alarm. This is because water flow in the sprinkler system can be seen on multiple switches for the same location. However, smoke and flame detectors are all individual alarms spaced far enough apart that each should be considered independent of each other.

An "Incipient Alarm" meets the intent of a "single fire alarm." A "pre-alarm" does not meet the intent of a "single fire alarm."

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 HU4.2 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.

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.

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

Background

Table H-1 Fire Areas are based on CP M-10, Fire Protection of Safe Shutdown Equipment. 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.

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.

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 HU4.2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

DCPP Basis Reference(s):

1. CP M-10, Fire Protection of Safe Shutdown Equipment
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 ISFSI PROTECTED AREA or PLANT PROTECTED AREA **not** extinguished within 60 minutes of the initial report, alarm or indication. (Note 1)	

Note 1: The SM/SEC/ED 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.

ISFSI PROTECTED AREA - Areas within the ISFSI to which access is strictly controlled in accordance with the station's Security Plan.

PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

Basis:

<u>ERO Decision Making Information</u>

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

In addition to a FIRE addressed by EAL HU4.1 or HU4.2, a FIRE within the PLANT PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety. This basis extends to a FIRE occurring within the Protected Area of the 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.

<u>Background</u>

<u>None</u>

DCPP 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 ISFSI PROTECTED AREA or PLANT PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.	

Mode Applicability:

All

Definition(s):

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

ISFSI PROTECTED AREA - Areas within the ISFSI to which access is strictly controlled in accordance with the station's Security Plan.

PLANT PROTECTED AREA - Areas to which access is strictly controlled in accordance with the station's Security Plan.

Note: The DCPP Independent Spent Fuel Storage Installation (ISFSI) has its own ISFSI PROTECTED AREA separate from the PLANT PROTECTED AREA.

Basis:

ERO Decision Making Information

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 ISFSI or PLANT PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department, for DCPP this is normally CalFire), 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 (engages in firefighting efforts or is needed to engage in firefighting efforts) because the fire is beyond the capability of the Fire Brigade (for DCPP, this is the DCPP Fire Department) 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.

Background

None

DCPP 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 Rooms/Areas	
Room/Area	**Mode(s)**
Auxiliary Building – 115' - BASTs	2, 3, 4
Auxiliary Building – 100' – BA Pumps	2, 3, 4
Auxiliary Building – 85' – Aux Control Board	2, 3, 4
Auxiliary Building – 64' – BART Tank area	2, 3, 4
Area H (below Control Room) – 100' 480V Bus area/rooms	3, 4

Mode Applicability:

2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown

Definition(s):

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

Basis:

<u>ERO Decision Making Information</u>

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. Specifically, the identified rooms are those where an activity must be performed to borate to cold shutdown, isolate accumulators or cooldown using RHR.

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

An 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 SM/SAC/ED 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.

This EAL does not apply to firefighting activities that automatically or manually activate a fire suppression system in an area. Such events are classified per IC HU4 - Fire.

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

<u>Background</u>

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

An 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.

NOTE: *IC HA5 mode applicability has been limited to the applicable modes identified in Table H-2 Safe Operation & Shutdown Rooms/Areas. If due to plant operating procedure or plant configuration changes, the applicable plant modes specified in Table H-2 are changed, a corresponding change to Attachment 3 'Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases' and to IC HA5 mode applicability is required.*

DCCP Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Rooms/Areas Table 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 requiring plant control to be transferred from the Control Room to the Hot Shutdown Panel area.

Mode Applicability:

All

Definition(s):

None

Basis:

ERO Decision Making Information

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

The Shift Manager (SM) determines if the Control Room requires evacuation and entry into OP AP-8A. 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. OP AP-8A Control Room Inaccessibility – Establishing Hot Standby and OP AP-8B Control Room Inaccessibility – Hot Standby to Cold Shutdown provides the instructions establishing plant control from outside the Control Room (Ref. 1, 2).

Inability to establish plant control from outside the Control Room escalates this event to a Site Area Emergency per EAL HS6.1.

Escalation of the emergency classification level would be via IC HS6.

Background

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.

DCPP Basis Reference(s):

1. OP AP-8A Control Room Inaccessibility – Establishing Hot Standby

2. OP AP-8B Control Room Inaccessibility – Hot Standby to Cold Shutdown
3. OP AP-34.5.1 Fire Response – Cable Spreading Room (FA 7-A)
4. OP AP-34.5.3 Fire Response – Control Room (CR-1)
5. 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 Hot Shutdown Panel area.

AND

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

- Reactivity (Modes 1, 2 and 3 **only**)
- Core Cooling
- RCS heat removal

Note 1: The SM/SEC/ED 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:

ERO Decision Making Information

The Shift Manager (SM) determines if the Control Room requires evacuation per OP AP-8A. 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. OP AP-8A Control Room Inaccessibility – Establishing Hot Standby and OP AP-8B Control Room Inaccessibility – Hot Standby to Cold Shutdown, provides the instructions establishing plant control from outside the Control Room (Ref. 1, 2).

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on SM/SEC/ED judgment. The SM/SEC/ED 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). The 15 minute clock starts once plant control has been transferred to the Hot Shutdown Area (OP AP-8A Attachment 4 480V Bus Alignment and Appendix F Electrical System Actions).

Physical control of key safety functions by manipulation of controls is **not** required to verify control, rather, it is sufficient that control transfer is successful (i.e. light indication of applicable equipment).

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

Background

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 shut down the reactor and maintain it shutdown), RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink).

Hot Shutdown Panel (HSDP) indications for Reactivity, Core Cooling and RCS Heat Removal:

- Reactivity
  - Gamma Metrics indicators (NI-53 & NI-54)
- Core Cooling
  - Pressurizer Liquid Temperature (TI-453B)
  - Pressurizer Pressure (PI-455B)
  - RCS WR Pressure (PI-406 at Dedicated Shutdown Panel)
  - RCS Temperatures (Loop 1 at Dedicated Shutdown Panel)
- RCS heat removal
  - AFW Flow Indicators (FI-165 through 168)
  - AFW Pump discharge pressures (PI-51B through 53B)
  - SG WR Levels (LI-501 through 504)
  - SG Pressures (PI-514, 524, 534, 544)

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.

DCPP Basis Reference(s):

1. OP AP-8A Control Room Inaccessibility – Establishing Hot Standby
2. OP AP-8B Control Room Inaccessibility – Hot Standby to Cold Shutdown
3. OP AP-34.5.1 Fire Response – Cable Spreading Room (FA 7-A)
4. OP AP-34.5.3 Fire Response – Control Room (CR-1)
5. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SM/SEC/ED Judgment

Initiating Condition: Other conditions existing that in the judgment of the SM/SEC/ED warrant declaration of a UE.

EAL:

HU7.1	**Unusual Event**
Other conditions exist which in the judgment of the SM/SEC/ED 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:

ERO Decision Making Information

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

Background

The SM/SEC/ED is the designated onsite individual having the responsibility and authority for implementing the DCPP Radiological Emergency Response Plan. The operations Shift Manager (SM) initially acts in the capacity of the SEC/ED and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the SM/SEC/ED, 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.

Refer to CP M-10 Fire Protection of Safe Shutdown Equipment for a list of SAFETY SYSTEMS.

DCPP Basis Reference(s):

1. NEI 99-01 HU7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SM/SEC/ED Judgment

Initiating Condition: Other conditions exist that in the judgment of the SM/SEC/ED warrant declaration of an Alert.

EAL:

HA7.1	**Alert**
Other conditions exist which, in the judgment of the SM/SEC/ED, 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):

EPA PROTECTIVE ACTION GUIDELINES (EPA PAG) - The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs requires DCPP to recommend protective actions for the general public to offsite planning agencies.

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

OWNER CONTROLLED AREA (OCA) - For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan.

SECURITY EVENT - Any incident representing an attempted, threatened, or actual breach of the security system or reduction of the operational effectiveness of that system. A security event can result in either a SECURITY CONDITION or HOSTILE ACTION.

Basis:

ERO Decision Making Information

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

Background

The SM/SEC/ED is the designated onsite individual having the responsibility and authority for implementing the DCPP Radiological Emergency Response Plan. The operations Shift Manager (SM) initially acts in the capacity of the SEC/ED and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the SM/SEC/ED, 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.

DCPP Basis Reference(s):

1. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SM/SEC/ED Judgment

Initiating Condition: Other conditions existing that in the judgment of the SM/SEC/ED warrant declaration of a Site Area Emergency.

EAL:

HS7.1 **Site Area Emergency**
Other conditions exist which in the judgment of the SM/SEC/ED 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):

EPA PROTECTIVE ACTION GUIDELINES (EPA PAG) - The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs requires DCPP to recommend protective actions for the general public to offsite planning agencies.

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

OWNER CONTROLLED AREA (OCA) - For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan.

SITE BOUNDARY - As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous/Liquid Effluent Release Points.

Basis:

<u>ERO Decision Making Information</u>

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

<u>Background</u>

The SM/SEC/ED is the designated onsite individual having the responsibility and authority for implementing the DCPP Radiological Emergency Response Plan. The operations Shift Manager (SM) initially acts in the capacity of the SEC/ED and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the SM/SEC/ED, 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.

DCPP Basis Reference(s):

1. NEI 99-01 HS7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SM/SEC/ED Judgment

Initiating Condition: Other conditions exist which in the judgment of the SM/SEC/ED warrant declaration of a General Emergency.

EAL:

HG7.1	**General Emergency**
Other conditions exist which in the judgment of the SM/SEC/ED 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):

EPA PROTECTIVE ACTION GUIDELINES (EPA PAG) - The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs requires DCPP to recommend protective actions for the general public to offsite planning agencies.

HOSTILE ACTION - An act toward DCPP 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 DCPP. 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.

OWNER CONTROLLED AREA (OCA) - For purpose of HOSTILE ACTION classifications, in accordance with this EAL scheme, the OCA is defined as the same area and boundary contained in the DCPP Security and Safeguards Contingency Plan.

Basis:

<u>ERO Decision Making Information</u>

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

Releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the SITE BOUNDARY.

Background

The SM/SEC/ED is the designated onsite individual having the responsibility and authority for implementing the DCPP Radiological Emergency Response Plan. The operations Shift Manager (SM) initially acts in the capacity of the SEC/ED and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the SM/SEC/ED, 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.

DCPP Basis Reference(s):

1. 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 Vital AC Power

Loss of vital 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 vital buses.

2. Loss of Vital DC Power

Loss of vital 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 125 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. RTS Failure

This subcategory includes events related to failure of the Reactor Trip System (RTS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RTS 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 RTS 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.

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.

Category: S – System Malfunction

Subcategory: 1 – Loss of Vital AC Power

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

EAL:

SU1.1 Unusual Event

Loss of **all** offsite AC power capability, Table S-1, to Unit 1 or Unit 2 vital 4.16KV buses F, G and H for ≥ 15 minutes. (Note 1)

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

Table S-1 AC Power Capability		
	Unit 1	**Unit 2**
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• Aux XFMR 1-2 fed from the Main Generator • DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• Aux XFMR 2-2 fed from the Main Generator • DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

Mode Applicability:

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

Definition(s):

None

Basis:

ERO Decision Making Information

For emergency classification purposes, "capability" means that, whether or not the buses are actually powered from it, an AC power source(s) can be aligned to the vital buses within 15 minutes:

- By a clear procedure path,
  and
- Breakers and equipment are readily available to power up the bus within the allotted time frame.

The vital 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant (see figure below).

This EAL addresses a prolonged (greater than 15 minutes) 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.

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

When filling out the ENF form, this event can be Unit 1, Unit 2 or Unit 1 and 2.

Escalation of the emergency classification level would be via IC SA1.

Background

Unit 1(2) 4.16KV buses F, G and H are the emergency (vital) buses. Each bus has three sources of power.

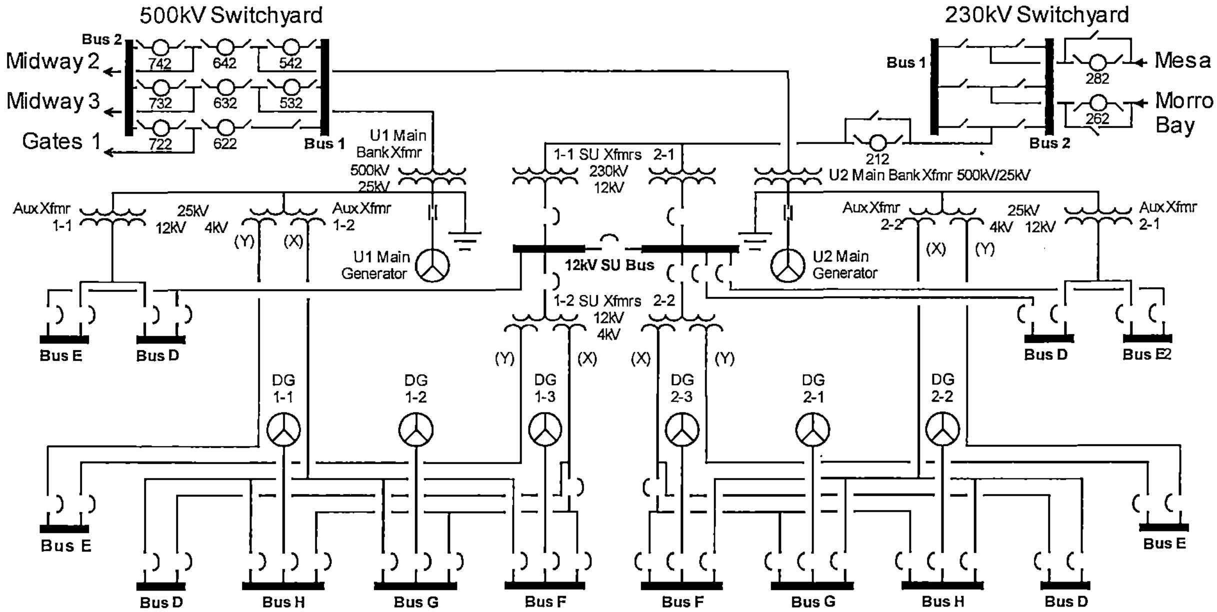
One offsite source is from the 230KV switchyard via 12KV startup transformer 1-1 (2-1) to the 4.16KV startup transformer 1-2 (2-2).

If the 1-1- or 2-1 transformer is unavailable, the other unit's #1 transformer (2-1 or 1-1) can be used to supply the startup bus through the startup bus cross tie breaker.

Another method to obtain offsite power is by back feeding the vital buses from the 500KV switchyard through the main transformer to the 4.16KV unit auxiliary transformer 1-2 (2-2). This is normally only done post-trip when 500KV power is available.

During normal operations, vital bus power is supplied from onsite by the main generator via the 4.16KV unit auxiliary transformer 1-2 (2-2). In addition, each units vital buses F, G and H have an onsite emergency diesel generator which can supply electrical power to its associated bus automatically in the event that the preferred source becomes unavailable (ref. 1-6).

DCPP Electrical Distribution System



DCPP Basis Reference(s):

1. UFSAR, Section 8.2.2
2. UFSAR, Section 8.3.1.6
3. OP AP SD-1, Loss of AC Power
4. OP AP-2, Loss of Offsite Power
5. OP J-2:V, Backfeeding the Unit From the 500kV System
6. ECA-0.0, Loss of All Vital AC Power
7. NEI 99-01 SU1

Category: S – System Malfunction

Subcategory: 1 – Loss of Vital AC Power

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

EAL:

SA1.1 Alert

AC power capability, Table S-1, to Unit 1 or Unit 2 vital 4.16KV buses F, G and H reduced to a single power source for ≥ 15 minutes. (Note 1)

AND

A failure of that single power source will result in loss of **all** AC power to SAFETY SYSTEMS.

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

Table S-1 AC Power Capability

	Unit 1	Unit 2
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• Aux XFMR 1-2 fed from the Main Generator • DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• Aux XFMR 2-2 fed from the Main Generator • DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

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:

ERO Decision Making Information

For emergency classification purposes, "capability" means that, whether or not the buses are actually powered from it, an AC power source(s) can be aligned to the vital buses within 15 minutes:

- By a clear procedure path,
  and
- Breakers and equipment are readily available to power up the bus within the allotted time frame.

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

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

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

When filling out the ENF form, this event can be Unit 1, Unit 2 or Unit 1 and 2.

Escalation of the emergency classification level would be via IC SS1.

Background

The vital 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant (see figure below).Unit 1(2) 4.16KV buses F, G and H are the emergency (vital) buses. Each bus has three sources of power.

One offsite source is from the 230KV switchyard via 12KV startup transformer 1-1 (2-1) to the 4.16KV startup transformer 1-2 (2-2).

If the 1-1- or 2-1 transformer is unavailable, the other unit's #1 transformer (2-1 or 1-1) can be used to supply the startup bus through the startup bus cross tie breaker.

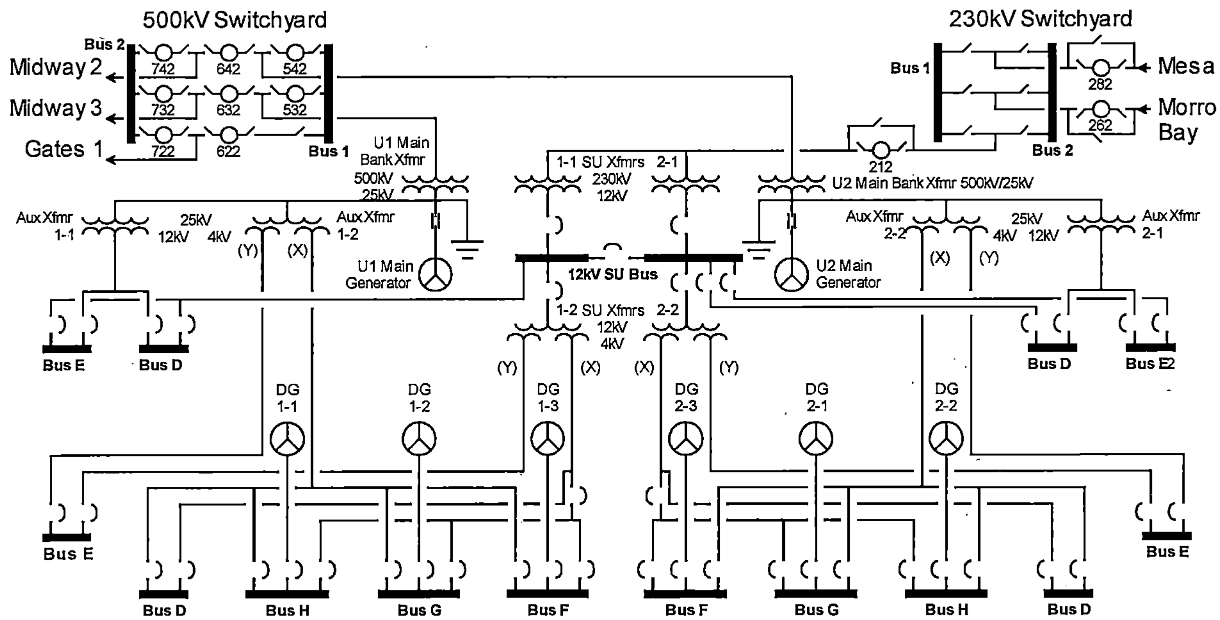
Another method to obtain offsite power is by back feeding the vital buses from the 500KV switchyard through the main transformer to the 4.16KV unit auxiliary transformer 1-2 (2-2). This is normally only done post-trip when 500KV power is available.

During normal operations, vital bus power is supplied from onsite by the main generator via the 4.16KV unit auxiliary transformer 1-2 (2-2). In addition, each units vital buses F, G and H have an onsite emergency diesel generator which can supply electrical power to its associated bus automatically in the event that the preferred source becomes unavailable (ref. 1-6).

If the capability of a second source of emergency bus power is not restored within 15 minutes, an Alert is declared under this EAL.

Refer to CP M-10 Fire Protection of Safe Shutdown Equipment for a list of SAFETY SYSTEMS.

DCPP Electrical Distribution System



DCPP Basis Reference(s):

1. UFSAR, Section 8.2.2
2. UFSAR, Section 8.3.1.6
3. OP AP SD-1, Loss of AC Power
4. OP AP-2, Loss of Offsite Power
5. OP J-2:V, Backfeeding the Unit From the 500kV System
6. ECA-0.0, Loss of All Vital AC Power
7. NEI 99-01 SA1

Category: S – System Malfunction

Subcategory: 1 – Loss of Vital AC Power

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

EAL:

SS1.1 Site Area Emergency

Loss of **all** offsite and **all** onsite AC power capability, Table S-1, to Unit 1 or Unit 2 vital 4.16KV buses F, G and H for ≥ 15 minutes. (Note 1)

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

Table S-1 AC Power Capability		
	Unit 1	**Unit 2**
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• Aux XFMR 1-2 fed from the Main Generator • DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• Aux XFMR 2-2 fed from the Main Generator • DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

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:

<u>ERO Decision Making Information</u>

For emergency classification purposes, “capability” means that, whether or not the buses are actually powered from it, an AC power source(s) can be aligned to the vital buses within 15 minutes:

- By a clear procedure path,
  and
- Breakers and equipment are readily available to power up the bus within the allotted time frame.

The 15-minute interval begins when both offsite and onsite AC power capability are lost.

This EAL addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

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

When filling out the ENF form, this event can be Unit 1, Unit 2 or Unit 1 and 2.

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

Background

The vital 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant (see figure below).

Unit 1(2) 4.16KV buses F, G and H are the emergency (vital) buses. Each bus has three sources of power.

One offsite source is from the 230KV switchyard via 12KV startup transformer 1-1 (2-1) to the 4.16KV startup transformer 1-2 (2-2).

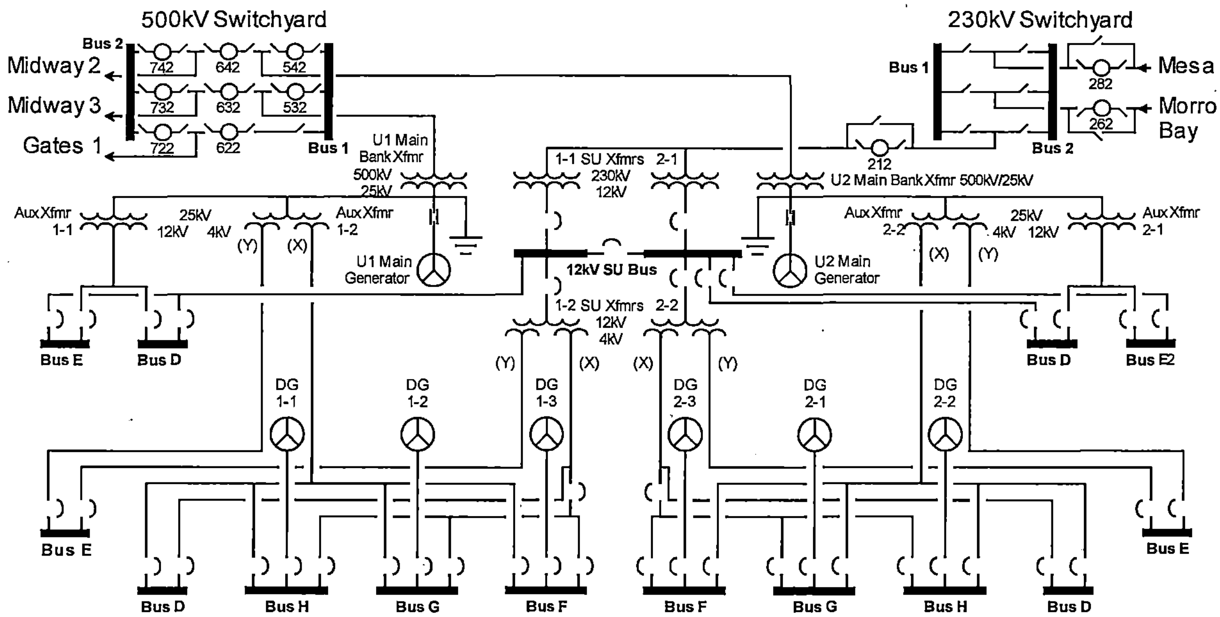
If the 1-1- or 2-1 transformer is unavailable, the other unit's #1 transformer (2-1 or 1-1) can be used to supply the startup bus through the startup bus cross tie breaker.

Another method to obtain offsite power is by back feeding the vital buses from the 500KV switchyard through the main transformer to the 4.16KV unit auxiliary transformer 1-2 (2-2). This is normally only done post-trip when 500KV power is available.

During normal operations, vital bus power is supplied from onsite by the main generator via the 4.16KV unit auxiliary transformer 1-2 (2-2). In addition, each units vital buses F, G and H have an onsite emergency diesel generator which can supply electrical power to its associated bus automatically in the event that the preferred source becomes unavailable (ref. 1-6).

ATTACHMENT 1
EAL Bases

DCPP Electrical Distribution System



DCPP Basis Reference(s):

1. UFSAR, Section 8.2.2
2. UFSAR, Section 8.3.1.6
3. OP AP SD-1, Loss of AC Power
4. OP AP-2, Loss of Offsite Power
5. OP J-2:V, Backfeeding the Unit From the 500kV System
6. ECA-0.0, Loss of All Vital AC Power
7. NEI 99-01 SS1

Category: S –System Malfunction

Subcategory: 1 – Loss of Vital AC Power

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

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power capability, Table S-1, to Unit 1 or Unit 2 vital 4.16KV buses F, G and H.

AND EITHER:

- Restoration of at least one 4.16KV vital bus in < 4 hours is **not** likely. (Note 1)
- CSFST Core Cooling RED path conditions met.

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

Table S-1 AC Power Capability

	Unit 1	Unit 2
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• Aux XFMR 1-2 fed from the Main Generator • DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• Aux XFMR 2-2 fed from the Main Generator • DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

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:

ERO Decision Making Information

This EAL is indicated by the extended loss of all offsite and onsite AC power capability to 4.16KV vital buses F, G and H either for greater then the DCPP Station Blackout (SBO) coping analysis time (4 hrs.) (ref. 1) 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. 2).

For emergency classification purposes, "capability" means that, whether or not the buses are actually powered from it, an AC power source(s) can be aligned to the vital buses within 15 minutes:

- By a clear procedure path,
  and
- Breakers and equipment are readily available to power up the bus within the allotted time frame.

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.

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.

When filling out the ENF form, this event can be Unit 1, Unit 2 or Unit 1 and 2.

Background

The vital 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant (see figure below).

Unit 1(2) 4.16KV buses F, G and H are the emergency (vital) buses. Each bus has three sources of power.

One offsite source is from the 230KV switchyard via 12KV startup transformer 1-1 (2-1) to the 4.16KV startup transformer 1-2 (2-2).

If the 1-1 or 2-1 transformer is unavailable, the other unit's #1 transformer (2-1 or 1-1) can be used to supply the startup bus through the startup bus cross tie breaker.

Another method to obtain offsite power is by back feeding the vital buses from the 500KV switchyard through the main transformer to the 4.16KV unit auxiliary transformer 1-2 (2-2). This is normally only done post-trip when 500KV power is available.

During normal operations, vital bus power is supplied from onsite by the main generator via the 4.16KV unit auxiliary transformer 1-2 (2-2). In addition, each units vital buses F, G and H have an onsite emergency diesel generator which can supply electrical power to its associated bus automatically in the event that the preferred source becomes unavailable (ref. 3-8).

Four hours is the station blackout coping time (ref 1).

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on SM/SEC/ED 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 (ref.2). Specifically, Core Cooling RED Path conditions exist if either:

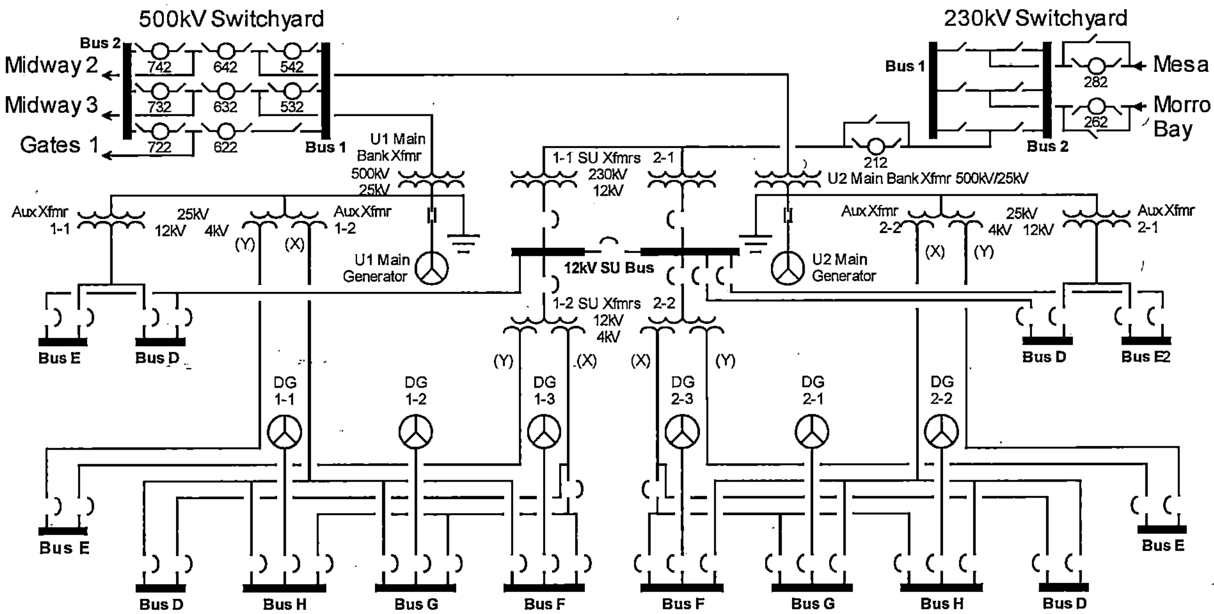
- Core exit TCs are reading greater than or equal to 1200°F, or
- Core exit TCs are reading greater than or equal to 700°F with RCS subcooling less than or equal to 20°F, and RVLIS full range indication is less than or equal 32%.

This EAL 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.

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.

DCPP Electrical Distribution System

ATTACHMENT 1
EAL Bases



DCPP Basis Reference(s):

1. DCM T-42, Station Blackout
2. F-0, Critical Safety Function Status Trees Attachment 2, Core Cooling
3. UFSAR, Section 8.2.2
4. UFSAR, Section 8.3.1.6
5. OP AP SD-1, Loss of AC Power
6. OP AP-2, Loss of Offsite Power
7. OP J-2:V, Backfeeding the Unit From the 500kV System
8. ECA-0.0, Loss of All Vital AC Power
9. 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** 125 VDC power based on battery bus voltage indications < 105 VDC on **all** Unit 1 or Unit 2 vital DC buses for ≥ 15 minutes. (Note 1)	

Note 1: The SM/SEC/ED 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):

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:

ERO Decision Making Information

Minimum battery voltage of 105 VDC is the voltage below which supplied loads may not function (ref. 1, 3, 4).

This IC addresses a loss of vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

Background

The vital 125 VDC system consists of three independent networks per unit. Each network contains the following components:

- Battery
- Battery charger
- Standby battery charger to allow maintenance and/or testing

- Distribution panels
- Ground detector

The vital 125 VDC batteries and distribution panels are located in Area "H" of the 115 feet elevation of the Auxiliary Building. There are a total of three batteries per unit, 11(21), 12(22), and 13(23). The batteries are sized to provide sufficient power to operate the associated DC loads for the time necessary to safely shut down the unit, should a 480-VAC source to one or more battery chargers be unavailable (ref. 1, 2, 3).

DCPP Basis Reference(s):

1. ECA-0.0, Loss of All Vital AC Power
2. UFSAR, Section 8.3.2.2.2
3. OP AP-23, Loss of Vital DC Bus
4. Notification 50804190 DC Bus Voltage Trigger for EALs
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 capability, Table S-1, to Unit 1 or Unit 2 vital 4.16KV buses F, G and H for ≥ 15 minutes.

AND

Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **all** Unit 1 or Unit 2 vital DC buses for ≥ 15 minutes.

(Note 1)

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

Table S-1 AC Power Capability

	Unit 1	Unit 2
Offsite	• Startup XFMR 1-2 via Startup XFMR 1-1 • Startup XFMR 1-2 via Startup XFMR 2-1 • Aux XFMR 1-2 backfed via Main XFMR	• Startup XFMR 2-2 via Startup XFMR 1-1 • Startup XFMR 2-2 via Startup XFMR 2-1 • Aux XFMR 2-2 backfed via Main XFMR
Onsite	• Aux XFMR 1-2 fed from the Main Generator • DG 1-1 – Bus H • DG 1-2 – Bus G • DG 1-3 – Bus F • Other Unit via Startup Bus X-Tie	• Aux XFMR 2-2 fed from the Main Generator • DG 2-2 – Bus H • DG 2-1 – Bus G • DG 2-3 – Bus F • Other Unit via Startup Bus X-Tie

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:

ERO Decision Making Information

For emergency classification purposes, "capability" means that, whether or not the buses are actually powered from it, an AC power source(s) can be aligned to the vital buses within 15 minutes:

- By a clear procedure path,
  and
- Breakers and equipment are readily available to power up the bus within the allotted time frame.

Minimum battery voltage of 105 VDC is the voltage below which supplied loads may not function (ref.6, 8, 9).

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

Background

This EAL is indicated by the loss of all offsite and onsite vital AC power capability to 4.16KV vital buses F, G and H 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 vital 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant (see figure below).

Unit 1(2) 4.16KV buses F, G and H are the emergency (vital) buses. Each bus has three sources of power.

One offsite source is from the 230KV switchyard via 12KV startup transformer 1-1 (2-1) to the 4.16KV startup transformer 1-2 (2-2).

Another method to obtain offsite power is by back feeding the vital buses from the 500KV switchyard through the main transformer to the 4.16KV unit auxiliary transformer 1-2 (2-2). This is normally only done post-trip when 500KV power is available.

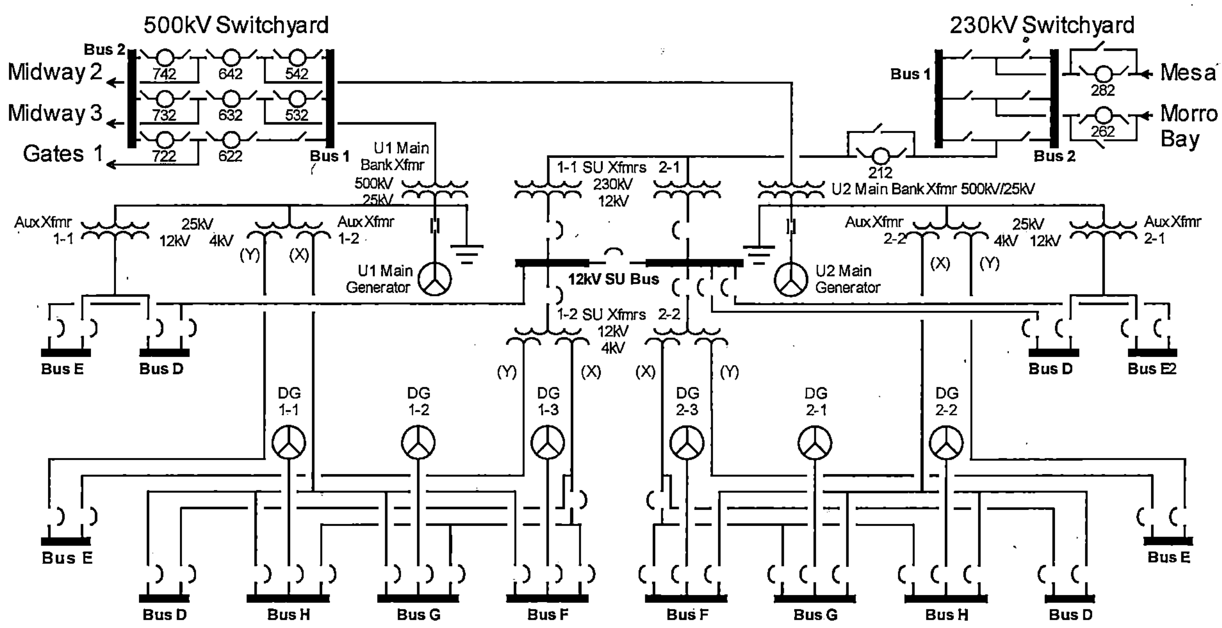
During normal operations, vital bus power is supplied from onsite by the main generator via the 4.16KV unit auxiliary transformer 1-2 (2-2). In addition, each units vital buses F, G and H have an onsite emergency diesel generator which can supply electrical power to its associated bus automatically in the event that the preferred source becomes unavailable (ref. 1-6).

The vital 125 VDC system consists of three independent networks per unit. Each network contains the following components:

- Battery
- Battery charger
- Standby battery charger to allow maintenance and/or testing
- Distribution panels
- Ground detector

The vital 125 VDC batteries and distribution panels are located in Area "H" of the 115 feet elevation of the Auxiliary Building. A total of three batteries per unit, 11(21), 12(22), and 13(23) are supplied for Units 1 and 2. The batteries are sized to provide sufficient power to operate the associated DC loads for the time necessary to safely shut down the unit, should a 480-VAC source to one or more battery chargers be unavailable (ref. 7, 8).

DCPP Electrical Distribution System



DCPP Basis Reference(s):

1. UFSAR, Section 8.2.2
2. UFSAR, Section 8.3.1.6
3. OP AP SD-1, Loss of AC Power
4. OP AP-2, Loss of Offsite Power
5. OP J-2:V, Backfeeding the Unit From the 500kV System
6. ECA-0.0, Loss of All Vital AC Power

7. UFSAR, Section 8.3.2.2.2
8. OP AP-23, Loss of Vital DC Bus
9. Notification 50804190 DC Bus Voltage Trigger for EALs
10. 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 minutes. (Note 1)	

Note 1: The SM/SEC/ED 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 or emergency feed flow in at least one SG

Mode Applicability:

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

Definition(s):

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

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

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

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

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

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

Basis:

<u>ERO Decision Making Information</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 Computer (PPC), ERFDS and SPDS serve as a redundant compensatory indicators which may be utilized in lieu of normal Control Room indicators (ref. 1).

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.

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.

<u>Background</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.

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.

DCPP Basis Reference(s):

1. UFSAR Section 7.5 Safety-Related Display 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 minutes. (Note 1)

AND

Any significant transient is in progress, Table S-3.

Note 1: The SM/SEC/ED 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 or emergency feed flow in at least one SG

Table S-3 Significant Transients
• Reactor trip • Runback ≥ 25% thermal power • Electrical load rejection > 25% electrical load • ECCS actuation

Mode Applicability:

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

Definition(s):

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

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

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

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

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

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

Basis:

ERO Decision Making Information

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 (PPC), ERFDS and SPDS serve 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.

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.

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

Background

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.

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.

DCPP Basis Reference(s):

1. UFSAR Section 7.5 Safety-Related Display Instrumentation
2. NEI 99-01 SA2

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification permissible limits.

EAL:

SU4.1	**Unusual Event**
RCS activity > Technical Specification Section 3.4.16 permissible limits.	

Mode Applicability:

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

Definition(s):

None

Basis:

ERO Decision Making Information

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications.

This EAL would be met if TS 3.4.16 Required Action C.1 (place plant in Mode 3 in 6 hours) or C.2 (place plant in Mode 5 in 36 hours) were not met.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

Background

The specific iodine activity is limited to 1.0 μCi/gm Dose Equivalent I-131. However, operation with iodine specific activity levels greater than the limit is permissible, if the activity levels do not exceed 60.0 μCi/gm Dose Equivalent I-131, for more than 48 hours.

The specific Xe-133 activity is limited to ≤ 600 μCi/gm Dose Equivalent XE-133 (ref 1).

With the Dose Equivalent I-131 greater than the LCO limit of 1 μCi/gm, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is < 60.0 μCi/gm. Dose Equivalent I-131 must be restored to within limits within 48 hours. This is acceptable since it is expected that, if there were an iodine spike, the normal RCS iodine concentration (≤ 1 μCi/gm) would be restored within this time period (ref 2).

This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

DCPP Basis Reference(s):

1. DCPP Technical Specifications section 3.4.16 RCS Specific Activity
2. DCPP Technical Specifications Basis section 3.4.16 RCS Specific Activity
3. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits.

EAL:

SU4.2	**Unusual Event**
With letdown in service, procedurally directed letdown dose point radiation > 3 R/hr.	

Mode Applicability:

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

Definition(s):

None

Basis:

ERO Decision Making Information

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.

Background

Initial indication of Fuel Clad degradation can be determined by measuring the external radiation dose rate at a distance of one foot from the center of the letdown line in the letdown heat exchanger room using the technique described in Attachment 7.1 of EP RB-14A, Initial Detection of Core Damage. An external radiation dose rate exceeding 3 R/hr indicates Fuel Clad degradation greater than Technical Specification allowable limits. This value was determined by ratioing 15 R/hr which corresponds to coolant activity at 300 µCi/gm to the Technical Specification LCO coolant activity of 60 µCi/gm which includes iodine spike (see EAL SU4.1), or 15 R/hr x 60/300 = 3 R/hr (ref 1, 2, 3).

DCPP Basis Reference(s):

1. EP RB-14A, Initial Detection of Core Damage
2. DCPP Technical Specifications section 3.4.16 RCS Specific Activity
3. PG&E Calculation EP 95-02 Rev. 0, Letdown Heat Exchanger Rom Dose Rates Corresponding to EP G-1, Alert No. 2 RCS Activity
4. 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 minutes. **OR** RCS identified leakage > 25 gpm for ≥ 15 minutes. **OR** Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 minutes. (Note 1)

Note 1: The SM/SEC/ED 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):

RCS LEAKAGE – RCS leakage shall be:

a. Identified Leakage

   1. Leakage, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leak off), that is captured and conducted to collection systems or a sump or collecting tank;
   2. 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;
   3. Reactor Coolant System (RCS) leakage through a steam generator to the secondary system (primary to secondary leakage).

b. Unidentified Leakage

   All leakage (except RCP seal water injection or leak off) that is not identified leakage.

c. Pressure Boundary Leakage

   Leakage (except primary to secondary leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

f. 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).

Basis:

ERO Decision Making Information

These conditions apply to leakage into the containment, a secondary-side system (e.g., steam generator tube leakage) or a location outside of containment.

The first and second EAL conditions are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications) (ref. 2).

The third condition addresses an RCS mass loss caused by an UNISOLABLE leak through an interfacing system (ref. 3, 4, 5).

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

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.

Below is a summary of classification guidance for steam generator tube leaks:

	Affected SG is FAULTED Outside of Containment?	
Primary-to-Secondary 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

Background

Manual or computer-based methods of performing an RCS inventory balance are normally used to determine RCS LEAKAGE.

STP R-10C, Reactor Coolant System Water Inventory Balance, is performed to determine the source and flow rate of the leakage. (ref. 1).

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 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.

DCPP Basis Reference(s):

1. STP R-10C, Reactor Coolant System Water Inventory Balance
2. DCPP Technical Specifications Definitions section 1.1
3. UFSAR Section 5.2.7 Reactor Coolant Pressure Boundary Leakage Detection System
4. UFSAR Section 5.2.9 Leakage Prediction From Primary Coolant Sources Outside Containment
5. OP AP-1, Excessive Reactor Coolant System Leakage
6. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 6 – RTS 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** RTS setpoint is exceeded. **AND** A subsequent automatic trip or manual trip action taken at the control room panels (CC1, VB2 or VB5) 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:

ERO Decision Making Information

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the control room panels (CC1, VB2 or VB5):

- Reactor trip switches (CC1 and VB2)
- Deenergization of 480V Buses 13D and 13E (23D and 23E) at the Control Room vertical board (VB5)

Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as locally opening the reactor trip beakers, local deenergization of 480V Buses 13D and 13E, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 4).

Following any automatic RTS trip signal, E-0 (ref. 2) and FR-S.1 (ref. 4) prescribe insertion of redundant manual trip signals to back up the automatic RTS 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 RTS 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.

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".

Should a reactor trip signal be generated as a result of plant work (e.g., RTS 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 RTS fails to automatically shut down the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the trip failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

Background

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Trip System (RTS) trip function. A reactor trip is automatically initiated by the RTS when certain continuously monitored parameters exceed predetermined setpoints (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 RTS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2, 3, 4).

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 RTS 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 shut down 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 shut down 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.

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.

DCPP Basis Reference(s):

1. DCPP Technical Specifications Section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. E-0 Reactor Trip or Safety Injection
3. F-0 Critical Safety Function Status Trees - Subcriticality
4. FR-S.1 Response to Nuclear Power Generation/ATWS
5. UFSAR Section 7.6.2.3 ATWS Mitigation Actuation Circuitry (AMSAC)
6 NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RTS 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 control room panels (CC1, VB2 or VB5) 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):

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:

ERO Decision Making Information

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the control room panels (CC1, VB2 or VB5):

- Reactor trip switches (CC1 and VB2)
- Deenergization of 480V Buses 13D and 13E (23D and 23E) at the Control Room vertical board (VB5)

Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as locally opening the reactor trip beakers, local deenergization of 480V Buses 13D and 13E (23D and 23E), 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).

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.

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".

Should a reactor trip signal be generated as a result of plant work (e.g., RTS 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 RTS fails to automatically shut down the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the trip failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

Background

This EAL addresses a failure of a manually initiated trip in the absence of having exceeded an automatic RTS 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 a manual 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 RTS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2, 3, 4).

This IC addresses a failure of the RTS 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 shut down 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 shut down 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.

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.

DCPP Basis Reference(s):

1. DCPP Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. E-0 Reactor Trip or Safety Injection
3. F-0 Critical Safety Function Status Trees - Subcriticality
4. FR-S.1 Response to Nuclear Power Generation/ATWS
5. UFSAR Section 7.6.2.3 ATWS Mitigation Actuation Circuitry (AMSAC)
6. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 2 – RTS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are **not** successful in shutting down the reactor.

EAL:

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 control room panels (CC1, VB2 or VB5) 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):

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:

ERO Decision Making Information

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the control room panels (CC1, VB2 or VB5):

- Reactor trip switches (CC1 and VB2)
- Deenergization of 480V Buses 13D and 13E (23D and 23E) at the Control Room vertical board (VB5)

Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as locally opening the reactor trip beakers, local deenergization of 480V Buses 13D and 13E (23D and 23E), 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).

Escalation of this event to a Site Area Emergency would be under EAL SS6.1 or SM/SEC/ED judgment.

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.

Background

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).

On the power range scale 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).

This IC addresses a failure of the RTS 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 shut down the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RTS.

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 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".

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. 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.

DCPP Basis Reference(s):

1. DCPP Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. E-0 Reactor Trip or Safety Injection
3. F-0 Critical Safety Function Status Trees - Subcriticality
4. FR-S.1 Response to Nuclear Power Generation/ATWS
5. UFSAR Section 7.6.2.3 ATWS Mitigation Actuation Circuitry (AMSAC)
6. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 2 – RTS 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 conditions met.
- CSFST Heat Sink RED path conditions met.
  AND
  Bleed and feed criteria met.

Mode Applicability:

1 - Power Operation

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:

ERO Decision Making Information

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.

Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as local deenergization of 480V Buses 13D and 13E (23D and 23E), 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, 4).

Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED path conditions being met (ref. 2).

Indication of inability to adequately remove heat from the RCS is manifested by CSFST Heat Sink RED path conditions being met in combination with bleed and feed criteria being met (ref. 3).

Heat Sink RED path conditions exist if:

- All SG narrow range levels less are than 15%, AND
- Less than 435 total AFW available to feed the SGs

Bleed and feed criteria are met when:

- Wide range level in any three (3) SGs is less than 18% [26%] AND
- All narrow range SG levels are less than 15% [25%].

Parenthetical values are used during Adverse Containment Conditions.

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

Background

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.

On the power range scale 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, 4).

This IC addresses a failure of the RTS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, all subsequent operator actions to manually shut down 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 shut down the reactor.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees – Attachment 1, Subcriticality
2. F-0 Critical Safety Function Status Tress – Attachment 2, Core Cooling
3. F-0 Critical Safety Function Status Tress – Attachment 3, Heat Sink
4. FR-S.1 Response to Nuclear Power Generation/ATWS
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 offsite communication methods.

OR

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

Table S-4 Communication Methods			
System	**Onsite**	**Offsite**	**NRC**
Unit 1, Unit 2 and TSC Radio Consoles	X	X	
DCPP Telephone System (PBX)	X	X	X
Portable radio equipment (handie-talkies)	X		
Operations Radio System	X	X	
Security Radio Systems	X		
CAS and SAS Consoles	X	X	X
Fire Radio System	X		
Hot Shutdown Panel Radio Consoles	X	X	X
Public Address System	X		
NRC FTS			X
Mobile radios	X		
Satellite phones	X	X	X
Direct line (ATL) to the County and State OES		X	

Mode Applicability:

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

Definition(s):

None

Basis:

ERO Decision Making Information

Onsite, offsite and NRC communications include one or more of the systems listed in Table S-4 (ref. 1, 2, 3).

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

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 offsite response organizations (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.).

Background

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 county EOCs.

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

DCPP Basis Reference(s):

1. UFSAR, Section 9.5.2
2. Emergency Plan Section 7.2 Communications Equipment
3. AR PK15-23, Communications
4. 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 minutes of a VALID containment isolation signal. (Note 1)

OR

Containment pressure ≥ 22 psig with < one full train of containment depressurization equipment operating per design for ≥ 15 minutes. (Notes 1, 9)

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

Note 9: One Containment Spray pump and two CFCUs 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:

ERO Decision Making Information

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 on 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. In order for a penetration to be considered isolated, a minimum of one valve in the flow path must be closed. 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. 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 or restore equipment that may not have automatically started or actuated as required, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment sprays) are either lost or performing in a degraded manner.

This event would escalate to a Site Area Emergency in accordance with IC FS1 if there were a concurrent loss or potential loss of either the Fuel Clad or RCS fission product barriers.

Background

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases requirement. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. The Refueling Water Storage Tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, Containment Spray, if needed, is transferred to the RHR Pumps and the Containment Spray Pumps are shut down(ref. 5).

The Containment Cooling System consists of two trains of Containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement. Each train, consisting of two Containment Fan Cooling Units (CFCU), is supplied with cooling water from a separate loop of Component Cooling Water (CCW). In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically if not already running (ref.5).

The Containment pressure setpoint (22 psig, ref. 1, 2, 3, 4) is the pressure at which the equipment should actuate and begin performing its function. The design basis accident analyses and evaluations assume the loss of one Containment Spray System train and one Containment Cooling System train (ref. 5). Consistent with the design requirement, "one full train of depressurization equipment" is therefore defined to be the availability of one train of each system. If less than this equipment is operating and Containment pressure is above the actuation setpoint, the threshold is met if the required equipment cannot be started within 15 minutes.

DCPP Basis Reference(s):

1. AR PK01-18, CONTMT SPRAY ACTUATION red
2. F-0 Critical Safety Function Status Trees - Attachment 6, Containment
3. FR-Z.1 Response to High Containment Pressure
4. Technical Specifications Table 3.3.2-1
5. Technical Specifications B3.6.6 Containment Spray and Cooling Systems
6. 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
• Tsunami
• Other events with similar hazard characteristics as determined by the SM/SEC/ED

Mode Applicability:

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

Definition(s):

DEGRADED PERFORMANCE – As applied to hazardous event thresholds, event damage significant enough to cause concern regarding the operability or reliability of the affected safety system train.

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.

TORNADO - A violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

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:

ERO Decision Making Information

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.

With respect to event damage caused by an equipment failure resulting in a FIRE or EXPLOSION, no emergency classification is required in response to a FIRE or EXPLOSION resulting from an equipment failure if the only safety system equipment affected by the event is that upon which the failure occurred. An emergency classification is required if a FIRE or EXPLOSION caused by an equipment failure damages safety system equipment that was otherwise functional or operable (i.e., equipment that was not the source/location of the failure). For example, if a FIRE or EXPLOSION resulting from the failure of a piece of safety system equipment causes damage to the other train of the affected safety system or another safety system, then an emergency declaration is required in accordance with this IC and EAL.

Escalation of the emergency classification level would be via IC FS1 or RS1.

Background

This condition represents an actual or potential substantial degradation of the level of safety of the plant. Due to this actual or potential substantial degradation, this condition can significantly reduce the margin to a loss or potential loss of a fission product barrier.

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.

DCPP Basis Reference(s):

1. NEI 99-01 SA9

Category F – Fission Product Barrier Degradation

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

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

A. 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 (CMT): 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 containment 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 DCPP 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 SM/SEC/ED 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

DCPP 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 SM/SEC/ED would have greater assurance that escalation to a General Emergency is less IMMINENT.

DCPP 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

DCPP 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. CMT Radiation / RCS Activity
D. CMT Integrity or Bypass
E. SM/SEC/ED 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 "CMT 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 (CMT) 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 conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B **Inadequate Heat Removal**	1. CSFST Core Cooling-RED path conditions met	1. CSFST Core Cooling-MAGENTA path conditions met 2. CSFST Heat Sink-RED path conditions met **AND** Bleed and feed criteria met	None	1. CSFST Heat Sink-RED path conditions met **AND** Bleed and feed criteria met	None	1. CSFST Core Cooling-RED path conditions met **AND** Restoration procedures **not** effective within 15 minutes (Note 1)
C **CMT Radiation / RCS Activity**	1. Containment radiation (RM-30 or RM-31) > 300 R/hr 2. Dose equivalent I-131 coolant activity > 300 µCi/gm	None	1. Containment radiation (RM-30 or RM-31) > 40 R/hr	None	None	1. Containment radiation (RM-30 or RM-31) > 5,000 R/hr
D **CMT Integrity or Bypass**	None	None	None	None	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on SM/SEC/ED determination • UNISOLABLE pathway from Containment to the environment exists 2. Indications of RCS LEAKAGE outside of Containment	1. CSFST Containment-RED path conditions met (≥ 47 psig) 2. Containment hydrogen concentration ≥ 4% 3. Containment pressure ≥ 22 psig with < one full train of depressurization equipment operating per design for ≥ 15 minutes (Note 1, 9)
E **SM/SEC /ED Judgment**	1. **Any** condition in the opinion of the SM/SEC/ED that indicates loss of the fuel clad barrier	1. **Any** condition in the opinion of the SM/SEC/ED that indicates potential loss of the fuel clad barrier	1. **Any** condition in the opinion of the SM/SEC/ED that indicates loss of the RCS barrier	1. **Any** condition in the opinion of the SM/SEC/ED that indicates potential loss of the RCS barrier	1. **Any** condition in the opinion of the SM/SEC/ED that indicates loss of the containment barrier	1. **Any** condition in the opinion of the SM/SEC/ED 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 conditions met.

Definition(s):

None

Basis:

ERO Decision Making Information

Core Cooling RED path conditions exist if either (ref. 1, 2):

- Core exit TCs are reading greater than or equal to 1200°F, or
- Core exit TCs are reading greater than or equal to 700°F with RCS subcooling less than or equal to 20°F and RVLIS full range indication is less than or equal 32% with no RCPs running

Background

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 dedicated SPDS display system. Some of the data is also available on the PPC, but the PPC is for information only (ref. 1, 2).

This reading indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees - Attachment 2, Core Cooling
2. FR-C.1 Response to Inadequate Core Cooling
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Core Cooling-MAGENTA path conditions met.

Definition(s):

None

Basis:

ERO Decision Making Information

Core Cooling MAGENTA path conditions exist if core exit subcooling margin is less than 20°F and any of the following (ref. 2, 3):

- RVLIS full range less than or equal to 32% with no RCPs running and less than 700°F, or
- Core exit TCs reading greater than or equal to 700°F with no RCPs running with greater than 32% RVLIS full range, or
- RVLIS dynamic range level less than or equal to the specified dynamic head value with one or more RCPs running, Table F-2

Table F-2 RVLIS Values

RVLIS	No. RCPs	Level
Full Range	None	32%
Dynamic Head	4	44%
	3	30%
	2	20%
	1	14%

Background

Critical Safety Function Status Tree (CSFST) Core Cooling-MAGENTA path indicates subcooling has been lost and that some fuel clad damage may potentially occur. The CSFSTs are normally monitored using the dedicated SPDS display system. Some of the data is also available on the PPC, but the PPC is for information only (ref. 1).

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees - Attachment 2, Core Cooling
2. FR-C.1 Response to Inadequate Core Cooling
3. FR-C.2 Response to Degraded Core Cooling
4. 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 conditions met.

   AND

   Bleed and feed criteria met.

Definition(s):

None

Basis:

ERO Decision Making Information

Heat Sink RED path conditions exist if:

- All SG narrow range levels less are than 15%, AND
- Less than 435 gpm total AFW available to feed the SGs

Bleed and feed criteria are met when:

- Wide range level in any three (3) SGs is less than 18% [26%] AND
- All narrow range SG levels are less than 15% [25%].

Parenthetical values are used during Adverse Containment Conditions.

In combination with RCS Potential Loss B.1, meeting this threshold results in a Site Area Emergency.

Background

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.

The CSFSTs are normally monitored using the dedicated SPDS display system. Some of the data is also available on the PPC, but the PPC is for information only (ref. 1).

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees - Attachment 3, Heat Sink
2. 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. CMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment radiation (RM-30 or RM-31) > 300 R/hr.

Definition(s):

None

Basis:

ERO Decision Making Information

Containment radiation monitor readings greater than 300 R/hr (ref. 1) indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the Containment. This value is higher than that specified for RCS barrier Loss C.1.

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.

Background

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors RM-30 and RM-31. These monitors provide indication in the Control Room on PAM 2 with a range of 1R/hr to 1E7 R/hr (ref. 1).

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 approximately 1.8% 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.

DCPP Basis Reference(s):

1. EP-CALC-DCPP-1602 Containment Radiation EAL Threshold Values
2. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.A

Barrier: Fuel Clad

Category: C. CMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. Dose equivalent I-131 coolant activity > 300 μCi/cc.

Definition(s):

None

Basis:

ERO Decision Making Information

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

This condition can be identified by either:

- RCS sample analysis
- EP RB-14A indications > 15 R/hr (ref. 1, 2)

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

Background

None

DCPP Basis Reference(s):

1. EP RB-14A Initial Detection of Fuel Cladding Damage
2. SPG-11 Obtaining the EP RB-14A Dose Rate
3. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.B

Barrier: Fuel Clad

Category: C. CMT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. SM/SEC/ED Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the SM/SEC/ED that indicates loss of the Fuel Clad barrier.

Definition(s):

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

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

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

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

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

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

Basis:

<u>ERO Decision Making Information</u>

The SM/SEC/ED 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.

- <u>IMMINENT barrier degradation</u> 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.
- <u>Barrier monitoring</u> 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.
- <u>Dominant accident sequences</u> lead to degradation of all fission product barriers and likely entry to the EOPs. The SM/SEC/ED 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 SM/SEC/ED in determining whether the Fuel Clad barrier is lost

Background

None

DCPP Basis Reference(s):

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

Barrier: Fuel Clad

Category: E. SM/SEC/ED Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the SM/SEC/ED that indicates potential loss of the Fuel Clad barrier.

Definition(s):

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

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:

ERO Decision Making Information

The SM/SEC/ED 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 SM/SEC/ED 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 SM/SEC/ED in determining whether the Fuel Clad barrier is potentially lost. The SM/SEC/ED should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Background

None

DCPP 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):

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.

RCS LEAKAGE – RCS leakage shall be:

a. Identified Leakage

   1. Leakage, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leak off), that is captured and conducted to collection systems or a sump or collecting tank;
   2. 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;
   3. Reactor Coolant System (RCS) leakage through a steam generator to the secondary system (primary to secondary leakage).

b. Unidentified Leakage

   All leakage (except RCP seal water injection or leak off) that is not identified leakage.

c. Pressure Boundary Leakage

   Leakage (except primary to secondary leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

g. 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).

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

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

NOTE: If an isolation valve could not be accessed due to local conditions (i.e. high radiation, temperature, etc.) then that would also make a leak unisolable, even though the inaccessible valve could isolate the leak.

Basis:

ERO Decision Making Information

A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED.

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.

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 will also be met.

Background

None

DCPP Basis Reference(s):

1. 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):

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.

RCS LEAKAGE – RCS leakage shall be:

a. Identified Leakage

   1. Leakage, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leak off), that is captured and conducted to collection systems or a sump or collecting tank;
   2. 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;
   3. Reactor Coolant System (RCS) leakage through a steam generator to the secondary system (primary to secondary leakage).

b. Unidentified Leakage

   All leakage (except RCP seal water injection or leak off) that is not identified leakage.

c. Pressure Boundary Leakage

   Leakage (except primary to secondary leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

h. 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).

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

NOTE: If an isolation valve could not be accessed due to local conditions (i.e. high radiation, temperature, etc.) then that would also make a leak unisolable, even though the inaccessible valve could isolate the leak.

Basis:

ERO Decision Making Information

The need to start an additional charging pump due to RCS LEAKAGE is an indication that the leak is in excess of charging pump capacity. This threshold is **not** met when an additional charging pump is started as prudent action. Rather, the threshold is met when an additional charging pump is started per conditions outlined in procedures OP AP-1 or OP AP-3, wherein RCS LEAKAGE exceeds capacity of a single charging pump with letdown isolated (ref. 1, 2).

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.

Background

None

DCPP Basis Reference(s):

1. OP AP-1 Excessive Reactor Coolant System Leakage
2. OP AP-3 Steam Generator Tube Failure
3. 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 conditions met.

Definition(s):

None

Basis:

ERO Decision Making Information

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 (PTS). PTS results from a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized).

Background

None

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees - Attachment 4, Integrity and 4a Limit A Curve
2. FR-P.1 Response to Imminent Pressurized Thermal Shock Condition
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.B

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Heat Sink-RED path conditions met.

   AND

   Bleed and feed criteria met.

Definition(s):

None

Basis:

ERO Decision Making Information

Heat Sink RED path conditions exist if:

- All SG narrow range levels less are than 15%, AND
- Less than 435 gpm total AFW available to feed the SGs

Bleed and feed criteria are met when:

- Wide range level in any three (3) SGs is less than 18% [26%] AND
- All narrow range SG levels are less than 15% [25%].

Parenthetical values are used during Adverse Containment Conditions.

In combination with RCS Potential Loss B.1, meeting this threshold results in a Site Area Emergency.

Background

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.

The CSFSTs are normally monitored using the dedicated SPDS display system. Some of the data is also available on the PPC, but the PPC is for information only (ref. 1).

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees - Attachment 3, Heat Sink
2. FR-H.1 Response to Loss of Secondary Heat Sink
3. NEI 99-01 Inadequate Heat Removal RCS Loss 2.B

Barrier: Reactor Coolant System

Category: C. CMT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment radiation (RM-30 or RM-31) > 40 R/hr.

Definition(s):

N/A

Basis:

ERO Decision Making Information

Containment radiation monitor readings greater than 40 R/hr (ref. 1) indicate the release of reactor coolant to the Containment.

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.

Background

The readings assume the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal coolant activity, with iodine spiking, discharged into containment (ref. 1).

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors RM-30 and RM-31. These monitors provide indication in the Control Room on PAM 2 with a range of 1R/hr to 1E7 R/hr (ref. 1).

DCPP Basis Reference(s):

1. EP-CALC-DCPP-1602 Containment Radiation EAL Threshold Values
2. NEI 99-01 CMT Radiation / RCS Activity RCS Loss 3.A

Barrier: Reactor Coolant System

Category: B. CMT Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. SM/SEC/ED Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the SM/SEC/ED that indicates loss of the RCS barrier.

Definition(s):

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

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:

ERO Decision Making Information

The SM/SEC/ED 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 SM/SEC/ED 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 SM/SEC/ED in determining whether the RCS Barrier is lost.

Background

None

DCPP Basis Reference(s):

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

Barrier: Reactor Coolant System

Category: E. SM/SEC/ED Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the SM/SEC/ED that indicates potential loss of the RCS barrier.

Definition(s):

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

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:

ERO Decision Making Information

The SM/SEC/ED 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 SM/SEC/ED 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 SM/SEC/ED in determining whether the RCS Barrier is potentially lost. The SM/SEC/ED should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Background

None

DCPP 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:

ERO Decision Making Information

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.

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 meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, gland seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category 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?	
Primary-to-Secondary 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

Background

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).

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.

There is no Potential Loss threshold associated with RCS or SG Tube Leakage.

DCPP 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 conditions met.

   AND

   Restoration procedures **not** effective within 15 minutes. (Note 1)

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

Definition(s):

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

Basis:

ERO Decision Making Information

The 15 minute clock starts upon entry into FR-C.1 Response to Inadequate Core Cooling (ref.2).

The restoration procedure is considered "effective" if core exit thermocouple readings are decreasing and/or if reactor vessel level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The SM/SEC/ED should escalate the emergency classification level as soon as it is determined that the procedure(s) will not be effective.

Background

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 dedicated SPDS display system (ref. 1). Some of the data is also available on the PPC, but the PPC is for information only

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).

A direct correlation to status trees can be made if the effectiveness of the restoration procedures is also evaluated. If core exit thermocouple (CET) readings are greater than 1,200°F, the Fuel Clad barrier is also lost (see Fuel Clad Loss B.1).

This condition represents an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. For this condition to occur, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation in a significant fraction of core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees - Attachment 2, Core Cooling
2. FR-C.1 Response to Inadequate Core Cooling
3. NEI 99-01 Inadequate Heat Removal Containment Potential Loss 2.A

Barrier: Containment

Category: C. CMT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: C. CMT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

1.	Containment radiation (RM-30 or RM-31) > 5,000 R/hr.

Definition(s):

None

Basis:

ERO Decision Making Information

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.

Background

Containment radiation monitor readings greater than 5,000 R/hr (ref. 1) indicate significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier (ref. 1).

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors RM-30 and RM-31. These monitors provide indication in the Control Room on PAM 2 with a range of 1R/hr to 1E7 R/hr (ref. 1).

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 associated 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.

DCPP Basis Reference(s):

1. EP-CALC-DCPP-1602 Containment Radiation EAL Threshold Values
2. NEI 99-01 CMT Radiation / RCS Activity Containment Potential Loss 3.A

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. Containment isolation is required.

   AND EITHER:

   - Containment integrity has been lost based on SM/SEC/ED determination.
   - UNISOLABLE pathway from containment to the environment exists.

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.

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

NOTE: If an isolation valve could not be accessed due to local conditions (i.e. high radiation, temperature, etc.) then that would also make a leak unisolable, even though the inaccessible valve could isolate the leak.

Basis:

ERO Decision Making Information

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold A.1.

First Bullet– 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).

Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the SM/SEC/ED 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.).

Second Bullet – 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.

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.

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.

<u>Background</u>

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.

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.

Refer to the middle piping run of Figure 1 on the following page. 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.

Refer to the top piping run of Figure 1 on the following page. 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.

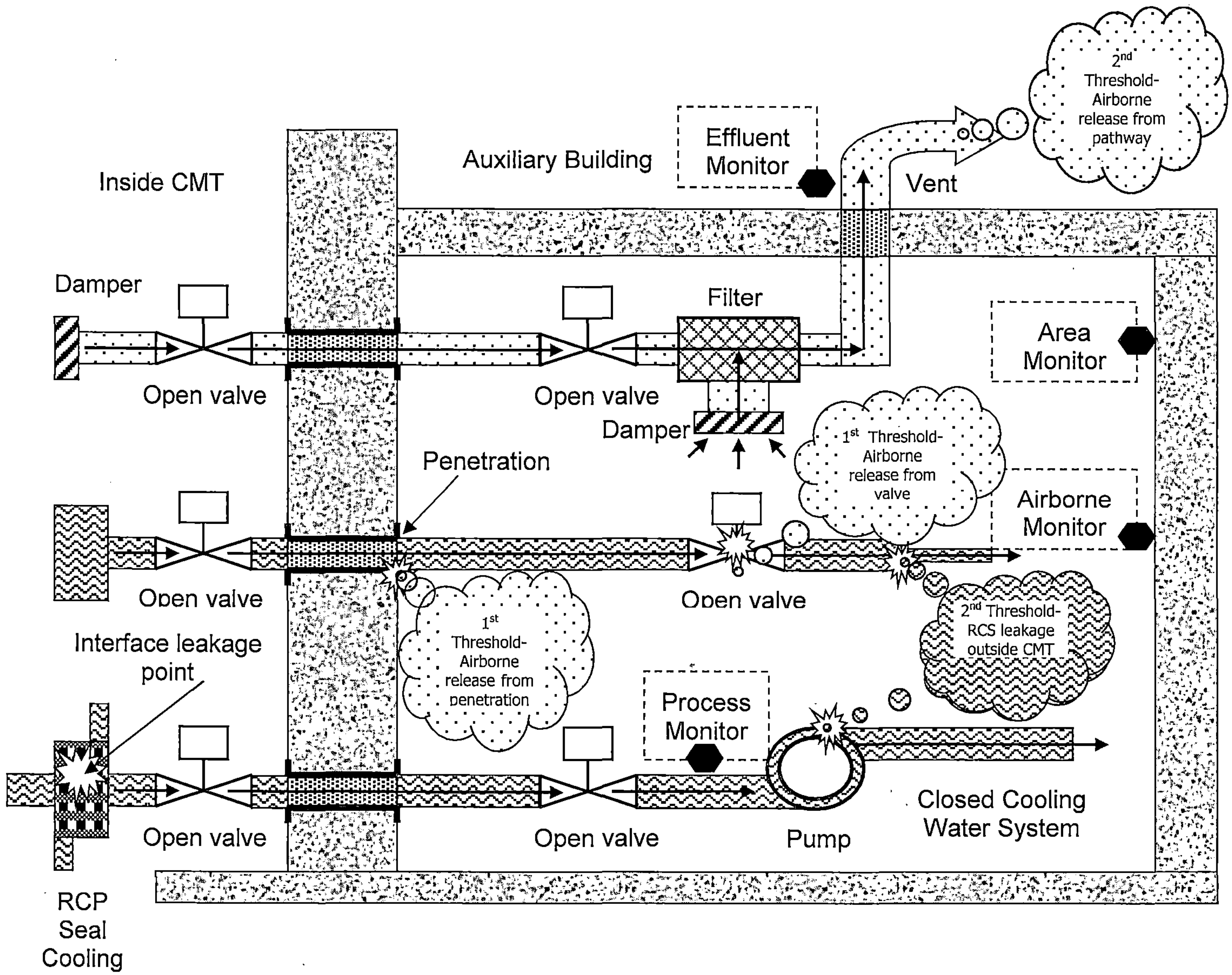
Refer to the bottom piping run of Figure 1 on the following page. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump developed a leak that allowed steam/water to enter the Auxiliary Building, then the second threshold would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause the first threshold to be met as well.

Following the leakage of RCS mass into containment and 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.

DCPP Basis Reference(s):

1. NEI 99-01 CMT Integrity or Bypass Containment Loss 4.A

Figure 1: Containment Integrity or Bypass Examples



Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

2.	Indications of RCS LEAKAGE outside of containment.

Definition(s):

RCS LEAKAGE – RCS leakage shall be:

a. Identified Leakage

 1. Leakage, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leak off), that is captured and conducted to collection systems or a sump or collecting tank;
 2. 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;
 3. Reactor Coolant System (RCS) leakage through a steam generator to the secondary system (primary to secondary leakage).

b. Unidentified Leakage

 All leakage (except RCP seal water injection or leak off) that is not identified leakage.

c. Pressure Boundary Leakage

 Leakage (except primary to secondary leakage) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

d. 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).

Basis:

<u>ERO Decision Making Information</u>

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
- PZR/RCS Loop sample lines

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.

The ECLs resulting from primary leakage outside containment (without a Fuel Clad challenge) are summarized below.

RCS LEAKAGE Outside Containment	**ECL**
< 25 gpm	No ECL
≥ 25 gpm – Charging Pump capacity	SU5.1
≥ Charging pump capacity	Site Area Emergency based on: RCS Potential Loss A.1 + Containment Loss D.2

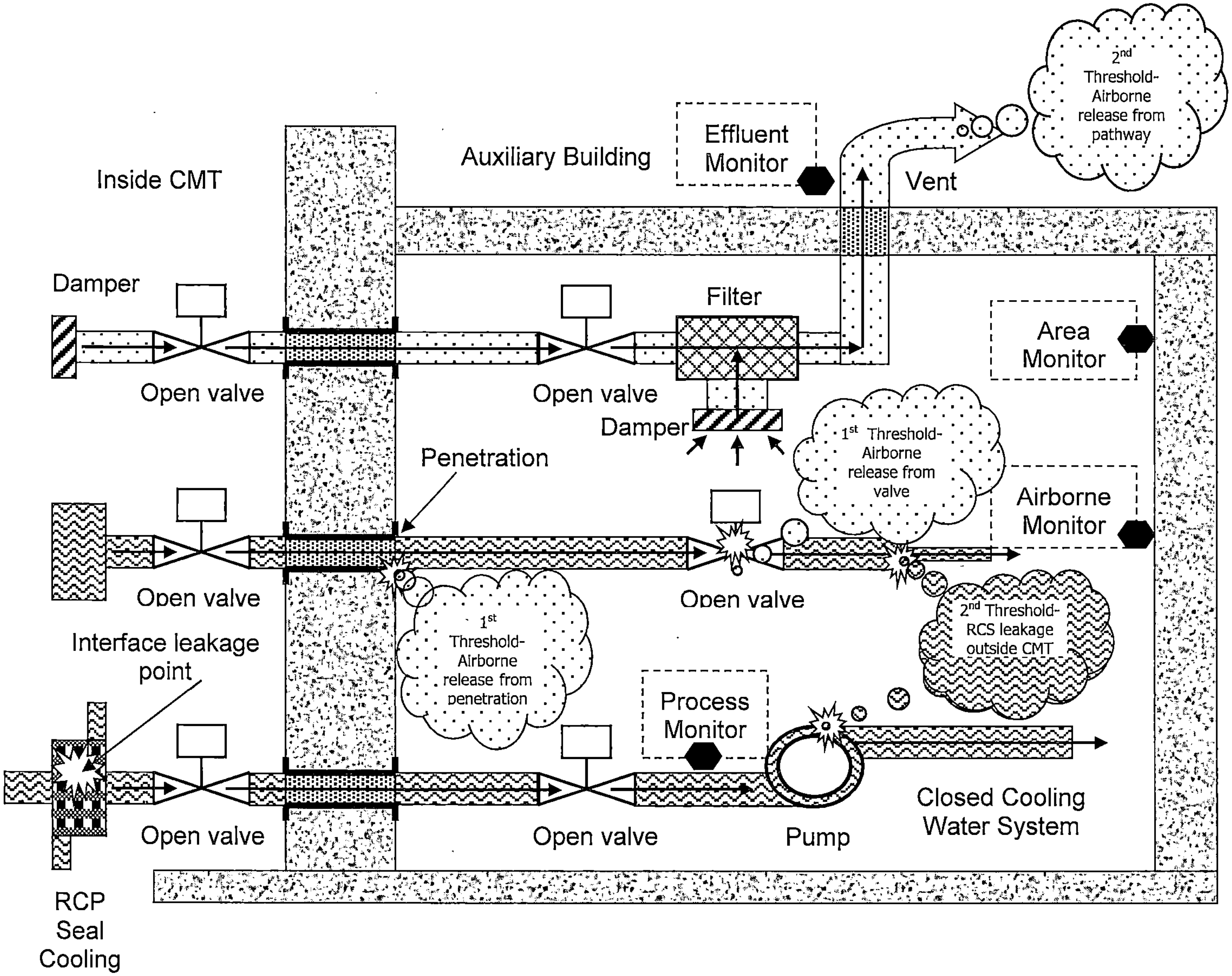
Background

Refer to the middle piping run of Figure 1 on the following page. 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.

DCPP Basis Reference(s):

1. ECA-1.2 LOCA Outside Containment
2. 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. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

1. CSFST Containment - RED path conditions met (≥ 47 psig).

Definition(s):

None

Basis:

ERO Decision Making Information

If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. As noted in the WOG SAMG and related DCPP implementation documents, 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.

Background

Critical Safety Function Status Tree (CSFST) Containment-RED path is entered if containment pressure is greater than or equal to 47 psig and represents an extreme challenge to safety function. The CSFSTs are normally monitored using the dedicated SPDS display system. Some of the data is also available on the PPC, but the PPC is for information only (ref. 1).

Forty-seven psig is the containment design pressure (ref. 1, 2) and is the pressure used to define CSFST Containment RED path conditions.

DCPP Basis Reference(s):

1. F-0 Critical Safety Function Status Trees – Containment, Attachment 5
2. FSAR Appendix 6.2D
3. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.A

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

2.	Containment hydrogen concentration ≥ 4%.

Definition(s):

None

Basis:

ERO Decision Making Information

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 flammability 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.

Background

After a LOCA, the containment atmosphere is a homogeneous mixture of steam, air, solid and gaseous fission products, hydrogen, and water droplets. During and following a LOCA, the hydrogen concentration in the containment results from radiolytic decomposition of water and metal-water reaction. If hydrogen concentration exceeds the lower flammability limit (4%) in an oxygen rich environment, a potentially explosive mixture exists. Operation of the Containment Hydrogen Recombiner with containment hydrogen concentrations greater than 4% could result in ignition of the hydrogen. If the combustible mixture ignites inside containment, loss of the containment barrier could occur. To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers must also have occurred. Since this threshold is also indicative of loss of both Fuel Clad and RCS barriers with the Potential Loss of the containment barrier, it therefore will likely warrant declaration of a General Emergency (ref. 1, 2, 3, 4).

Containment hydrogen concentration is indicated in the Control Room on ANR-82/ANR-83 PAM1, (range: 1 - 10%).

DCPP Basis Reference(s):

1. UFSAR Section 6.2.5 Combustible Gas Control In Containment
2. FR-Z.4 Response to High Containment Hydrogen Concentration
3. OP-H-9 INSIDE CONT H2 RECOMB SYSTEM
4. CA-3 Hydrogen Flammability in Containment
5. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.B

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

3. Containment pressure ≥ 22 psig.

AND

Less than one full train of containment depressurization equipment operating per design for ≥ 15 minutes. (Note 1, 9)

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

Note 9: One Containment Spray pump and two CFCUs comprise one full train of depressurization equipment.

Definition(s):

None

Basis:

ERO Decision Making Information

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 or restore equipment that may not have automatically started or actuated as required, if possible. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems (e.g., containment sprays, etc., but not including containment venting strategies) are either lost or performing in a degraded manner.

Background

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases requirement. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. The Refueling Water Storage Tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, Containment Spray, if needed, is transferred to the RHR Pumps and the Containment Spray Pumps are shut down (ref. 5).

The Containment Cooling System consists of two trains of Containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement. Each train, consisting of two Containment Fan Cooling Units (CFCU), is supplied with cooling water from a separate loop of Component Cooling Water (CCW). In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically if not already running (ref.5).

The Containment pressure setpoint (22 psig, ref. 1, 2, 3, 4) is the pressure at which the equipment should actuate and begin performing its function. The design basis accident analyses and evaluations assume the loss of one Containment Spray System train and one Containment Cooling System train (ref. 5). Consistent with the design requirement, "one full train of depressurization equipment" is therefore defined to be the availability of one train of each system. If less than this equipment is operating and Containment pressure is above the actuation setpoint, the threshold is met if the required equipment cannot be started within 15 minutes.

DCPP Basis Reference(s):

1. AR PK01-18, CONTMT SPRAY ACTUATION red
2. F-0 Critical Safety Function Status Trees - Attachment 6, Containment
3. FR-Z.1 Response to High Containment Pressure
4. Technical Specifications Table 3.3.2-1
5. Technical Specifications B3.6.6 Containment Spray and Cooling Systems
6. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.C

Barrier: Containment

Category: E. SM/SEC/ED Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the SM/SEC/ED that indicates loss of the containment barrier.

Definition(s):

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

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:

ERO Decision Making Information

The SM/SEC/ED 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 SM/SEC/ED 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 SM/SEC/ED in determining whether the Containment Barrier is lost.

Background

None

DCPP Basis Reference(s):

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

Barrier: Containment

Category: E. SM/SEC/ED Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the SM/SEC/ED that indicates potential loss of the containment barrier.

Definition(s):

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

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:

ERO Decision Making Information

The SM/SEC/ED 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 SM/SEC/ED 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 SM/SEC/ED in determining whether the Containment Barrier is lost.

Background

None

DCPP 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.*

Analysis

OP L-4, Normal Operation at Power (rev 89/73) was reviewed to determine if any actions are "necessary" to **maintain power operations**. Over reasonable periods of time (days vice months or years) there are no actions outside the Control Room that are required to be performed to maintain normal operations. Eventually, you would have to shut down if Technical Specification surveillance testing was not completed and you complied with the associated LCOs or based on consumable supplies being depleted. For the purpose of this table, no actions were determined to be required.

The following table lists the locations into which an operator may be dispatched in order **perform a normal plant cool down and shutdown**. The review was completed using the following procedures as the controlling documents:

OP L-4, Normal Operation at Power (rev 89/73) –

- Sections 6.3 (Instructions for Power Decreases from 100% to 50%)
- Section 6.4 (Instructions for Power Reduction From 50% to 20%)

OP L-5, Plant Cooldown From Minimum Load to Cold Shutdown (rev 100/83)

OP L-7, Plant Stabilization Following Reactor Trip (rev 24/22)

OP AP-25, Rapid Load Reduction or Shutdown (rev 25/12)

In addition, the Residual Heat Removal System is aligned per OP B-2:V "RHR - Place In Service" (rev 37/36) which was also used to conduct this review.

At DCPP, RCS Cooldown starts at OP L-5 step 6.2.3.m.

Each step in the controlling procedures was evaluated to determine if the action was performed in the Control Room or in the plant. Each in-plant action listed below was evaluated and a determination made whether or not the actions, if not performed, would prevent achieving cold shutdown. The following generic assumptions were applied:

- Steps involving optional degassing of the RCS were not selected since degassing the RCS is not required to reach cold shutdown.
- Steps involving supplying Auxiliary Steam were not selected since AFW can be used to reach cold shutdown if Condenser vacuum is lost.
- Steps involving Main Feed Water Pumps were not selected since AFW can be used to reach cold shutdown if Main Feed Water is not available.
- Steps that are stated as needed when entering an outage are disregarded, as they are optional and not mandatory for placing plant in Cold Shutdown.

Travel paths to the locations where the equipment is operated are not part of the determination of affected room/area, only the rooms/areas where the equipment is actually operated. Most locations can be reached via alternate travel paths if required due to a localized issue.

No assumption is made about which RHR Train is aligned for operation.

The minimum set of in-plant actions, associated locations, and operating modes to shut down and cool down the reactor are highlighted. The locations where those actions are performed comprise the rooms/areas to be included in EAL Tables R-2 and H-2. Specifically, the identified rooms are those where an activity must be performed to borate to cold shutdown, isolate accumulators or cooldown using RHR.

UFSAR Page 6.4-1 states "The DCPP control room, located at elevation 140 feet of the auxiliary building, is common to Unit 1 and Unit 2. The associated habitability systems provide for access and occupancy of the control room during normal operating conditions, radiological emergencies, hazardous chemical emergencies, and fire emergencies."

UFSAR Page 6.4-9 states "There are no offsite or onsite hazardous chemicals that would pose a credible threat to DCPP control room habitability. Therefore, engineered controls for the control room are not required to ensure habitability against a hazardous chemical threat and no amount of assumed unfiltered in-leakage is incorporated into PG&E's hazardous chemical assessment."

Control room habitability relative to area radiation levels is adequately bounded by EAL RA2.3.

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
OP L-4, Section 6.3: Instructions for Power Decreases from 100% to 50% **OP L-4, Section 6.4: Instructions for Power Reduction From 50% to 20%**				
OP L-4 6.3.3.b.2	Initiate RCS degassing as directed by chemistry PER OP B-1A:VIII, "Reactor Coolant System Degassing During a Plant Shutdown" OR OP B 1A:X, "CVCS - VCT Degassing."	Aux/100/various	1	No
OP L-4 6.3.3.l / 6.3.4.e	IF either cylinder heating steam pressure controller is in "MANUAL," THEN direct Turbine Building Watch to maintain cylinder heating pressure during the ramp PER OP C-3A:I, "Sealing Steam System - Place In Service."	TB/104	1	No
OP L-4 6.3.3.n	As power decreases, direct Nuclear Operators to adjust SGBD flows PER OP D-2:V, "Steam Generator Blowdown System - Place in Service."	TB/119	1	No
OP L-4 6.3.3.r.6 / 6.3.4.n.4	Direct operator in the field to open discharge vent to condenser valve on condensate pump that was just shut down: • CND PP 1-1: CND-1-31 • CND PP 1-2: CND-1-32 • CND PP 1-3: CND-1-33	TB/85	1	No
OP L-4 6.3.3.s / 6.3.4.i	WHEN less than 60% power, AND IF desired, THEN shut down one of the two running Circulating Water Pumps PER OP E-4:III, "Circulating Water System Shutdown and Clearing."	Intake	1	No
OP L-4 6.3.3.t.4 / 6.3.4.h.4	IF shutdown of MFW pump is required, THEN complete shutdown PER OP C-8:III, "Shutdown and Clearing of a Main Feed Water Pump."	TB/85	1	No
OP L-4 6.3.4.j	IF condenser is to be cleared upon reaching MODE 3, THEN consider realigning TDAFWP steam traps PER appropriate steps in OP L-5, "Plant Cooldown from Minimum Load to Cold Shutdown," section for "Secondary Plant Shutdown."	TB/104 AB/100/Pen	1	No

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
OP L-4 6.3.4.k	Direct Aux Watch to transfer aux steam supply to U2 PER OP K-5:IV "Auxiliary Steam System - Change Over to Alternate Supply of Steam."	AB/140	1	No
OP L-4 6.3.4.l	IF NOT already performed, THEN swap the Hydrazine injection points to the alternate alignment (downstream of FCV-232) per OP D-2:II, "Main Feed Water Chemical Injection System - Place in Service."	TB/85	1	No
OP L-4 6.3.4.m.2.a	IF unit is NOT being taken off line for OP L-8, "Separating From the Grid While Maintaining Reactor Power Between 17% and 30%"), THEN shut down No. 2 Heater Drip Pump PER OP C-7B:II, "No. 2 Heater Drip Pump Shutdown and Clearing."	TB/104, 85 & 70	1	No
OP L-4 6.3.4.t	On the MFW pump in service, locally place the HP and LP Stop Valves Drain control switch to the "OPEN" position to open the before-seat drains.	TB/85	1	No
OP L-5 Section 6.1.3: Power Decrease from 20% to MODE 3 with Normal Shutdown **OP L-5 Section 6.1.4: Power Decrease from 20% to MODE 3 with Planned Reactor Trip**				
OP L-5 6.1.3.d.2	IF Containment is to be entered, THEN Notify Chemistry to perform Containment air sampling.	Pen/100	1	No
OP L-5 6.1.3.m.13 / 6.1.4.t	Place AFW chemical injection in service PER OP D-2:I, "Auxiliary Feed Water Chemical Injection System - Place In Service."	AB/100	1	No
OP L-5 6.1.3.q	IMPLEMENT Section 10 to open FW-1-FCV-420 to prevent the FWH outlet relief from lifting.	TB	1/2/3	See step by step analysis of Section 10
OP L-5 6.1.3.s	IMPLEMENT step 11.5 for secondary shutdown.	TB	1/2/3	See step by step analysis of Section 11
OP L-5 6.1.3.w.8 / 6.1.4.u	Shut down both MFW pumps PER OP C-8:III, "Shutdown and Clearing of a Main Feed Water Pump."	TB	2/3	No
OP L-5 6.1.3.y.5 / 6.1.4.w	IF desired, THEN shut down the MG sets PER OP A-3:III, "Control Rod System - Shutdown & Clearing."	Area H/100	3	No

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
OP L-5 6.1.3.aa / 6.1.4.e.1	Initiate boration to the final concentration for the mode to which the plant is to be shut down PER one of the following • (Preferred) OP B-1A:XIX, "CVCS - Borate the RCS to Refueling Concentration" • (Alternate) OP B-1A:VII, Section 6.12, "Emergency Boration using CVCS-1-8104" • (Alternate) OP B-1A:VII, Section 6.3, "Routine Boration"	AB 100/115/East End (BAST & BA Pumps), 85' (Aux Control Board), 64' (BART Tank area)	2/3	Yes – basis of location is the ability to refill the BAST in order to have sufficient boric acid to reach CSD concentration. Cool down below 500°F requires 11000 gallons of boric acid be added (see step 6.2.3.d)
OP L-5 6.1.3.bb / 6.1.4.x	IF anticipated that the RCS will be opened and degassing of the RCS has not been started, THEN initiate degassing of the RCS to reduce H_2 concentration to 5 cc/kg or less PER OP B-1A:VIII, "CVCS - Reactor Coolant System Degassing During a Plant Shutdown."	AB/100	2/3	No
OP L-5 6.1.3.ee / 6.1.4.z	Maintenance to perform STP M-17B2, "Functional Test of Emergency DC Lighting System in Containment."	Various	1/2/3	No
OP L-5 6.1.3.gg / 6.1.4.bb	Ensure SGBD is maximized PER Chemistry direction and within the ability to control RCS temperature.	TB/119	1/2/3	No
OP L-5 section 6.2: MODE 3 to Ready for RHR Operation				
OP L-5 6.2.3.	Place the personnel airlock automatic leak rate monitor (ALRM) in manual PER STP M-8F1, "ALRM Leak Rate Testing of Personnel Air Lock Seals," to avoid nuisance alarms in the Control Room.	AB/140	3	No
OP L-5 6.2.3.e.2	Borate the RCS to meet STP R-19 COLD SHUTDOWN requirements.	AB 100/115/East End (BAST & BA Pumps), 85' (Aux Control Board), 64' (BART Tank area)	3/4	Yes – basis of location is the ability to refill the BAST in order to have sufficient boric acid to reach CSD concentration. (See Caution prior to step). TS 3.1.1
OP L-5 6.2.3.s	Close the accumulator isolation valve breakers • SI-1-8808A: breaker 52-1F-46 • SI-1-8808B: breaker 52-1G-07 • SI-1-8808C: breaker 52-1H-14	Area H/100/480V Buses	3/4	Yes – basis is that without closing Accumulator outlet valves, RCS pressure cannot go below ~650

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
	• SI-1-8808D: breaker 52-1G-05			psig (procedure does not address alternate actions) TS 3.5.1
OP L-5 6.2.3.y	WHEN desired, THEN secure the CRDM fans PER OP H-6:II, "CRDM Fans - Shutdown and Clearing."	Area H/100/480V Buses	3/4	No
OP L-5 6.2.3.cc.1.b	Disable BOTH SI Pumps PER OP O-32,"Control of Refueling Tags."	TB/119/4kV Vital Bus Rooms	3/4	No – ECG 8.6 violation, but does not prevent getting to CSD
OP L-5 6.2.3.cc.2.b	Disable ONE ECCS centrifugal charging pump PER OP O-32, "Control of Refueling Tags."	TB/119/4kV Vital Bus Rooms	3/4	No – ECG 8.6 violation, but does not prevent getting to CSD
OP L-5 section 6.3: Placing RHR in Service to CSD, Bubble in PZR				
OP L-5 6.3.3.b.4	Place RHR system in service PER OP B-2:V, "RHR-Place in Service During Plant Cooldown."		3/4	See step by step analysis of OP B-2:V
OP L-5 6.3.3.b.6	Place tags on RHR suction valves (RHR-1-8701 and RHR-1-8702) breakers PER OP O-32, "Control of Refueling Tags."	Area H/100/480V Buses	3/4	No – This is only a tag hanging step. Actual breaker manipulation is in OP B-2:V steps 6.2.12 / 6.3.12
OP L-5 6.3.3.d.1	Perform the following actions for CCP 1-3: Establish fire watch compensatory actions per ECG 8.1.	AB/73/CCP3 room	4	No – ECG 8.1 violation, but does not prevent getting to CSD
OP L-5 6.3.3.d.2	Perform the following actions for CCP 1-3: No more than one hour prior to reducing any WR RCS TCOLD to 283°F, make CCP 1-3 incapable of injecting.	TB/119/4kV Vital Bus Rooms	4	No – ECG 8.1 violation, but does not prevent getting to CSD
OP L-5 6.3.3.h	Hang the RCS Dilution Flow Path Boundary valve tags PER OP O-32, "Control of Refueling Tags."	AB/100	4	No
OP L-5 Section 10: Condensate System Long Recirc				
10.2	Ensure CLOSED FW-1-383, FCV-420 Downstream Isolation.	TB/85	3/4	No
10.3	Ensure CLOSED FW-1-384, FCV-420 Downstream Isolation Bypass	TB/85	3/4	No
10.4.1	Open FW-1-210, FW-1-211 Bypass.	TB/85	3/4	No
10.4.2	Open FW-1-211	TB/85	3/4	No
10.4.3	Close FW-1-210	TB/85	3/4	No

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
10.6	Ensure a minimum of four polisher vessels in service until long recirc is established.	TB/85	3/4	No
10.8.2	IF the temperature interlock is NOT made up, THEN contact Maintenance to open FW-1-FCV-420 by installing an air jumper with a 50 psig air supply connected to the vent side of SV1420.	TB/85	3/4	No
10.9	Coordinate with the Control Room and very slowly open FW-1-384 until the onset of FWH flashing, then throttle closed until it stops	TB/85	3/4	No
10.12	Slowly begin to open FW-1-383. If FWH flashing occurs, then throttle closed until it stops.	TB/85	3/4	No
10.14	Close FW-1-384.	TB/85	3/4	No
OP L-5 Section 11: Secondary System Shutdown				
11.2.2.a	Perform the following to prepare steam line drains for closing the MSIVs: Align valves for steam traps 1, 2, 3 and 5 steam line drains.	TB/104	3/4	No – If steam traps cannot be re-aligned declare AFW Pump 1 INOPERABLE.
11.2.2.b	Align AFW Pump 1-1 and Main Steam Traps 1, 2, 3 and 5 to the Outfall	TB/104 & Pen/100	3/4	No – If steam traps cannot be re-aligned declare AFW Pump 1 INOPERABLE.
11.2.3	Connect hoses for AFW chemical injection PER OP D-2:IV, "Adding Chemicals to Chemical Day Tanks-AFW System."	AB/100/AFW room	3/4	No
11.2.5	IF desired, THEN secure and clear a CWP PER OP E-4:III, "Circulating Water System Shutdown and Clearing."	Intake	3/4	No
11.2.7	IF the Main Generator is to be depressurized and purged, THEN warm up the CO2 vaporizer PER OP J-4C:III, "Generator Hydrogen System-Remove From Service."	TB/104	3/4	No
11.3	Just prior to separating from grid, drain MSR drain tanks and FW heaters PER OP C-7:III, "Condensate System - Shutdown and Layup."	TB/119	3/4	No
11.5.2	IF relatching the Main Turbine is	TB/140	3/4	No – If Cooldown

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
	needed to control plant cool down, THEN perform the following:[T35016] a. Close AIR-I-1-2489, Air Supply to the Air/Oil Relay. b. Isolate EH to the governor valves: • EH-1-518, for FCV-139 • EH-1-519, for FCV-140 • EH-1-520, for FCV-141 • EH-1-521, for FCV-142			control is an issue then MSIVs can be closed
11.5.3.b	Align the MSRs as necessary PER OP C-5:III, "Moisture Separator Reheaters - Shutdown."	TB/119 & 104	3/4	No
11.5.5	IF desired, THEN back feed the unit from 500kV PER OP J-2:V, "Back feeding the Unit from the 500kV System."	Various	3/4	No
11.5.7	Secure and drain SCCW PER OP J-4A:III, "Generator Stator Cooling Water-Shutdown and Draining."	TB/85	3/4	No
11.6.1	Depressurize and purge the Main Generator PER OP J-4C:III, "Generator Hydrogen System-Remove from Service."	TB/140 & 119	3/4	No
11.6.2	Secure SCW to exciter air coolers	TB/104	3/4	No
11.7.6	Remove polishers from service PER OP C-7C:II, "Condensate Polishing System-Remove Demineralizers from Service," as directed by the Secondary Foreman.	TB/85 & 104/Polishers	3/4	No
11.7.8	Open CND-1-506 to break vacuum.	TB/119	3/4	No
11.7.9	Maintenance to remove RM-15 and RM-15R from service.	TB/104	3/4	No
11.7.10	Secure gland steam and cylinder heating steam PER OP C-3A:III, "Sealing Steam System-Shutdown and Clearing.	TB/104 & 140	3/4	No
11.7.11	Secure condenser air removal PER OP C-6:III, "Condenser and Air Removal System-Shutdown and Clearing."	TB/104	3/4	No
11.7.12	Secure the following PER OP C-6C:II, "Condensate Air and Nitrogen Injection - Remove from Service:"	TB/119 & 140	3/4	No

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
	• N2 injection • Air injection			
11.7.14.b	Secure chemical injection PER OP D-2:II, "Main Feed Water Chemical Injection-Place in Service."	TB/85	3/4	No
11.7.15	Isolate condensate reject PER OP C-7:III, "Condensate System-Shutdown and Layup" (LCV-12).	TB/85	3/4	No
11.11.1	Secure turning gear PER OP C-3:IV, "Main Unit Turbine-Turbine Shutdown."	TB/140	3/4	No
11.11.2	Shut down lube oil PER OP C-3B:III, "Lube Oil Distribution System-Shutdown and Clearing."	TB/85, 104 & 119	3/4	No
11.11.4	Shut down H2 Seal Oil System PER OP J-4B:II, "Hydrogen Seal Oil System-Shutdown and Drain."	TB/85	3/4	No
11.12	WHEN the RCS is at or below 350°F, THEN remove locking devices on the following SGBD throttle valves and open them to achieve maximum blowdown	TB/119 & AB/140	3/4	No
OP B-2:V: RHR - Place In Service				
6.1.5	Shift chemistry/radiation protection technician to sample RHR Loop 1-1 to determine RHR Loop 1-1 boron concentration.	AB/100/PSSS	4	No - Basis is the system is aligned for ECCS Mode from the RWST. In the event of an SI, we do not verify boron concentration prior to injecting.
6.1.10	Shift chemistry/radiation protection technician to sample RHR Loop 1-2 to determine RHR Loop 1-2 boron concentration.	AB/100/PSSS	4	No - Basis is the system is aligned for ECCS Mode from the RWST. In the event of an SI, we do not verify boron concentration prior to injecting.
6.1.13.b / 6.2.27 / 6.2.43	Open RHR-1-8734A, RHR System 1-1 Bypass to Letdown Heat Exchanger Inlet (85' Containment Penetration Area).	Pen/85	4	No - Basis is the system is aligned for ECCS Mode from the RWST. In the event of an SI, we do not verify boron concentration prior to injecting.
6.1.13.i	Chemistry to sample RHR Loop 1-1 at approximately 10 minute	AB /100/PSSS	4	No - Basis is the system is aligned for ECCS

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
	intervals until the boron concentration is equal to or greater than that in the RCS.			Mode from the RWST. In the event of an SI, we do not verify boron concentration prior to injecting.
6.1.13.l / 6.2.36.b	Close RHR-1-8734A, RHR System 1-1 Bypass to Letdown Heat Exchanger Inlet.	Pen/85	4	No
6.1.13.q / 6.2.36.a / 6.3.8	Open RHR-1-8734B, RHR System 1-2 Bypass to Letdown Heat Exchanger Inlet (85'Containment Penetration Area).	Pen/85	4	No - Basis is the system is aligned for ECCS Mode from the RWST. In the event of an SI, we do not verify boron concentration prior to injecting.
6.1.13.u	Chemistry to sample the RHR loop at approximately 10 minute intervals until the boron concentration of RHR loop 1-2 is equal to or greater than that in the RCS.	AB /100/PSSS	4	No - Basis is the system is aligned for ECCS Mode from the RWST. In the event of an SI, we do not verify boron concentration prior to injecting.
6.1.13.w / 6.2.18	Close RHR-1-8734B, RHR System 1-2 Bypass to Letdown Heat Exchanger Inlet.	Pen/85	4	No
6.2.9 / 6.3.9	Open RHR-1-8726A, RHR Heat Exchanger 1-1 Bypass (64' elevation Auxiliary Building).	AB/64/RHR pumps hallway	4	No – This keeps the RHR trains split but does not prevent cool down.
6.2.10 / 6.3.10	Open RHR-1-8726B, RHR Heat Exchanger 1-2 Bypass (64' elevation Auxiliary Building).	AB/64/RHR pumps hallway	4	No – This keeps the RHR trains split but does not prevent cool down.
6.2.12 / 6.3.12	Ensure CLOSED the breakers for the following valves: • 52-1F-31, MOV 8980 • 52-1G-25, MOV 8701 • 52-1H-19, MOV 8702	Area H/100/480V Buses	4	Yes – required to align RHR system
OP L-7, Plant Stabilization Following Reactor Trip				
6.5.2	IF a Circulating Water pump was tripped, THEN REFER TO OP E-4:III, Circulating Water System – Shutdown and Clearing, for cleanup actions.	Intake	3	No
6.5.3	IF no Circulating Water pump can be placed in service, THEN cool	TB/various	3	No

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
	down a hot condenser in accordance with AP-7, Attachment 1			
6.10.7	Align SG Blowdown via the Blowdown Tank per OP D-2:V for SG chemistry and RCS temperature control.	TB/119 & Pen/100	3	No
6.11.4	Condensate Polisher Beds aligned per Secondary Foreman direction.	TB/104/Polisher	3	No
6.12.2.i	Open FW-1-FCV-420	TB/104	3	No
6.12.2.j	Coordinate with the Control Room and very slowly OPEN FW-1-384	TB/85	3	No
6.12.2.k	Very slowly OPEN FW-1-383.	TB/85	3	No
6.12.2.l	Close FW-1-384	TB/85	3	No
6.13.2.b	Realign steam traps 1, 2, 3, and 5 / steam line drains	TB/104	3/4	No – If steam traps cannot be re-aligned declare AFW Pump 1 INOPERABLE.
6.13.2.c & d	Align AFW Pump 1-1 and Main Steam Traps 1, 2, 3 and 5 to the Outfall	TB/104 & Pen/100	3/4	No – If steam traps cannot be re-aligned declare AFW Pump 1 INOPERABLE.
6.13.3	Align Auxiliary and Gland Seal steam as desired per OP C-3A:I.	TB/104 & AB/100	3/4	No
6.14.2	If desired to control plant cool down, relatch the Main Turbine as follows: a. Close AIR-I-1-2489, Air Supply to the Air/Oil Relay. b. Isolate EH to the governor valves: • EH-1-518, for FCV-139 • EH-1-519, for FCV-140 • EH-1-520, for FCV-141 • EH-1-521, for FCV-142	TB/140	3/4	No – If cooldown control is an issue then MSIVs can be closed
6.15	IF desired, THEN back feed the unit from 500kV PER OP J-2:V, "Back feeding the Unit from the 500kV System."	Various	3/4	No
6.31	On the 4kV vital buses, reset dropped flags on undervoltage relays 27HFB1, 27HGB1 and 27HHB1.	TB/119/4kV Vital Bus Rooms	3/4	No
OP AP-25, Rapid Load Reduction or Shutdown				

Procedure and Step	Step Action	Building/ Elevation/Room	Mode	If action not performed, does this prevent cool down/ shut down?
7.a RNO e / 14.f.4 / 20.c.4	WHEN plant conditions permit, THEN swap Condensate Pump vents PER OP C-7A:I.	TB/85	1/2/3	No

Table R-2 & H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode(s)**
Auxiliary Building – 115' - BASTs	2, 3, 4
Auxiliary Building – 100' – BA Pumps	2, 3, 4
Auxiliary Building – 85' – Aux Control Board	2, 3, 4
Auxiliary Building – 64' – BART Tank area	2, 3, 4
Area H (below Control Room) – 100' 480V Bus area/rooms	3, 4