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February 17, 2016
L-16-020

10 CFR 50, Appendix E

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

SUBJECT:
Davis-Besse Nuclear Power Station, Unit No. 1
Docket No. 50-346, License No. NPF-3
Request for Licensing Action to Revise the Emergency Plan

In accordance with the provisions of 10 CFR 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," Section IV.B.2, FirstEnergy Nuclear Operating Company (FENOC) is requesting an amendment to revise the current Davis-Besse Nuclear Power Station, Unit No. 1 Emergency Plan emergency action level scheme to one based on Nuclear Energy Institute (NEI) 99-01, "Development of Emergency Action Level for Non-Passive Reactors," Revision 6.

An evaluation of the proposed amendment is provided as Enclosure A. Calculations referenced in the proposed amendment are provided as Enclosure B. FENOC is requesting the Nuclear Regulatory Commission (NRC) staff approval by March 1, 2017, with an implementation period of 120 days following issuance of the amendment.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 315-6810.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 17, 2016.

Sincerely,

Brian D. Boles

Enclosures:

A. Evaluation of Proposed License Amendment
B. EAL Calculations

cc: NRC Region III Administrator
NRC Project Manager
NRC Resident Inspector
Executive Director, Ohio Emergency Management Agency State of Ohio (NRC Liaison)
Utility Radiological Safety Board

Enclosure A
L-16-020

Evaluation of Proposed License Amendment
(697 Pages Follow)

Subject: Request to Adopt Emergency Action Level Scheme Pursuant to 99-01, Revision 6, “Development of Emergency Action Levels for Non-Passive Reactors”

1.0 SUMMARY DESCRIPTION

In accordance with the provisions of 10 CFR 50.90 and 10 CFR 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," Section IV.B, FirstEnergy Nuclear Operating Company (FENOC), is proposing a change to the Davis-Besse Nuclear Power Station (DBNPS) Emergency Plan by revising the emergency action level (EAL) scheme.

FENOC proposes to change the EALs from a scheme based on Revision 5 of Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," to a scheme based on Revision 6 of NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors." Such a change in scheme requires NRC approval prior to implementation. The proposed change would continue to meet the standards in 10 CFR 50.47(b)(4) and the requirements in Appendix E to 10 CFR 50.

2.0 DETAILED DESCRIPTION

2.1 Background of NEI 99-01, Revision 6

In November 2012, NEI published NEI 99-01, Revision 6. The NRC formally endorsed NEI 99-01, Revision 6 as documented in a letter to NEI dated March 28, 2013 (ADAMS Accession Number ML12346A463). NEI 99-01, Revision 6 addresses changes recommended by the NRC, along with enhancements identified by the industry.

NEI 99-01, Revision 6, represents the most recently accepted EAL methodology endorsed by the NRC.

2.2 Proposed Change to Current EAL Scheme

DBNPS currently uses an emergency classification scheme based on NEI 99-01, Revision 5, as approved by the NRC in a letter dated December 19, 2008 (ADAMS Accession Number ML083450120). FENOC requests approval to change the DBNPS EAL scheme basis to that described in NEI 99-01, Revision 6. Detailed descriptions of the proposed changes and supporting information are attached:

- Attachment 1 - Emergency Action Level (EAL) Bases Document provides the proposed DBNPS EALs and EAL technical bases.
- Attachment 2 – Emergency Action Level (EAL) Bases Document (Redline Version) provides a marked up version of the proposed DBNPS EALs and EAL technical bases. It identifies the marked-up changes to NEI 99-01, Revision 6, EAL Basis as incorporated into Attachment 1; it is provided for information purposes only. Changes to the NEI 99-01, Revision 6, IC and EAL language are captured in the EAL comparison matrix.
- Attachment 3 – Davis-Besse Nuclear Power Station NEI 99-01, Revision 6, EAL Comparison Matrix compares the proposed DBNPS EALs to the NEI 99-01, Revision 6, EALs and provides the differences.

- Attachment 4 – Emergency Action Level (EAL) Wallcharts provides the proposed wallcharts that incorporate the requested changes for ease of use of the proposed DBNPS EALs.

3.0 TECHNICAL EVALUATION

3.1 NEI 99-01, Revision 6 Evaluation

NEI 99-01, Revision 6, provides guidance to nuclear power plant operators for the development of a site-specific emergency classification scheme. 10 CFR 50.47(b)(4) stipulates that emergency plans include a standard emergency classification and action level scheme. This scheme is a fundamental component of an emergency plan, in that it provides the defined thresholds that will allow site personnel to rapidly implement a range of pre-planned emergency response measures. An emergency classification scheme also facilitates timely decision-making by an emergency response organization (ERO) concerning the implementation of precautionary or protective actions for the public.

The initiating conditions (ICs), EALs, and basis that comprise the proposed emergency classification scheme are described in DBNPS EAL technical bases document (Attachment 1). A comparison matrix (Attachment 3) was developed to compare the DBNPS EALs to the NEI 99-01, Revision 6 EALs. This matrix provides a tabular format of the ICs, mode applicability, and EAL threshold values in NEI 99-01, Revision 6, along with the proposed EALs. The comparison matrix also compares the proposed EALs in terms of differences and deviations from the NRC-endorsed guidance provided in NEI 99-01, Revision 6.

The comparison matrix summarizes an evaluation that determined if the proposed IC and EAL wording represents no change from the guidance, a difference from the guidance, or a deviation from the guidance contained in NEI 99-01, Revision 6. Items were determined to be differences or deviations based on the definitions provided in Regulatory Issue Summary (RIS) 2003-18, Supplement 2. The RIS defines an EAL difference and deviation as follows:

- A difference is an EAL change where the basis scheme guidance differs in wording but agrees in meaning and intent, such that a classification of an event would be the same, whether using the basis scheme guidance or the site-specific proposed EAL. Examples of differences include the use of site-specific terminology or administrative re-formatting of site-specific EALs.
- A deviation is an EAL change where the basis scheme guidance differs in wording and is altered in meaning or intent, such that a classification of the event could be different between the basis scheme guidance and the site-specific proposed EAL. Examples of deviations include the use of altered mode applicability, altering key words or time limits, or changing words of physical reference (protected area, safety-related equipment, etc.).

The evaluation determined that the proposed DBNPS specific ICs and EALs contain differences from the NEI 99-01, Revision 6, guidance, but do not contain deviations. The basis for each difference is included in Attachment 3.

3.2 Instrumentation Validation

NRC Information Notice (IN) 2013-01, "Emergency Action Level Thresholds Outside the Range of Radiation Monitors," is intended to inform licensees of the importance of having adequate procedures to properly evaluate changes to site procedures, equipment, and facilities for potential impact on the licensee's ability to maintain an effective emergency plan. Specifically, the IN informs licensees of issues that arose when radiation monitors were not properly evaluated in conjunction with changes made to EAL thresholds for emergency classifications. The NRC also alerted licensees to similar issues in IN 2005-19, "Effect of Plant Configuration Changes on the Emergency Plan."

Using the guidance from IN 2013-01, IN 2005-19, and NEI 99-01, Revision 6, Section 4.3 "Instrumentation Used for EALs," FENOC conducted reviews to verify the threshold values in the proposed EAL scheme could be accurately read on the instrumentation referenced in the EALs. The reviews concluded that the proposed EAL threshold values are within the requisite calibrated ranges of the instrumentation referenced in the EALs and they can be accurately read.

3.3 EAL Scheme Change Evaluation

10 CFR 50, Appendix E, Section IV.B.2 stipulates that a licensee desiring to change its entire EAL scheme shall submit an application for an amendment to its license and receive NRC approval before implementing the change.

The proposed changes to the EAL scheme to adopt the NEI 99-01, Revision 6, guidance, do not reduce the capability to meet the applicable emergency planning requirements established in 10 CFR 50.47 and 10 CFR 50, Appendix E.

The proposed changes to adopt the NEI 99-01, Revision 6, EAL scheme will continue to provide consistent emergency classifications. Changes to DBNPS's emergency plan and procedures resulting from implementation of the revised EALs will be evaluated in accordance with the requirements of 10 CFR 50.54(q), subsequent to NRC approval.

Accordingly, pursuant to the requirements of 10 CFR 50, Appendix E, Section IV.B.2, FENOC requests NRC review and approval of the proposed changes to the DBNPS EAL scheme in accordance with 10 CFR 50.90.

4.0 REGULATORY EVALUATION

4.1 Significant Hazards Consideration

FirstEnergy Nuclear Operating Company (FENOC), requests an amendment to the Facility Operating License for Davis-Besse Nuclear Power Station (DBNPS) to support the adoption of an emergency action level (EAL) scheme based on Nuclear Energy Institute (NEI) 99-01, Revision 6.

The proposed changes to DBNPS's EAL scheme does not reduce the capability to meet the emergency planning requirements established in 10 CFR 50.47 and 10 CFR 50, Appendix E. The proposed changes do not reduce the functionality, performance, or capability of DBNPS's Emergency Response Organization (ERO) to respond in mitigating the consequences of accidents. All DBNPS ERO functions will continue to be performed as required.

FENOC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

Response: No.

The proposed changes to DBNPS's EAL scheme to adopt the NRC-endorsed guidance in NEI 99-01, Revision 6, do not involve any physical changes to plant systems or equipment. The proposed changes do not alter any of the requirements of the technical specifications. The proposed changes do not modify any plant equipment and do not impact any failure modes that could lead to an accident. Additionally, the proposed changes do not impact the ability of structures, systems, or components (SSCs) to perform their intended safety functions in mitigating the consequences of an initiating event within the assumed acceptance limits.

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

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

Response: No.

The proposed changes to DBNPS's EAL scheme to adopt the NRC-endorsed guidance in NEI 99-01, Revision 6, do not involve any physical changes to plant systems or equipment. The proposed changes do not involve the addition of any new plant equipment. The proposed changes will not alter the design configuration, or method of operation of plant equipment beyond its normal functional capabilities. DBNPS functions will continue to be performed as required. The proposed changes do not create any new credible failure mechanisms, malfunctions, or accident initiators.

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

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

Response: No.

The proposed changes to DBNPS's EAL scheme to adopt the NRC-endorsed guidance in NEI 99-01, Revision 6, do not involve any physical changes to plant systems or equipment. Margins of safety are unaffected by the proposed changes. There are no changes being made to safety analysis assumptions, safety limits, or limiting safety system settings that would adversely affect plant safety as a result of the proposed EAL scheme change. The proposed change does not affect the technical specifications. There are no changes to environmental conditions of any of the SSC or the manner in which any SSC is operated. The applicable requirements of 10 CFR 50.47 and 10 CFR 50, Appendix E will continue to be met.

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

Based on the above, FENOC concludes that the proposed amendment 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.

4.2 Applicable Regulatory Requirements/Criteria

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

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

10 CFR 50 Appendix E, section IV, "Content of Emergency Plan," 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.

FENOC has determined that the proposed amendment maintains conformance with the regulatory requirements described above.

4.3 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the 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.

5.0 ENVIRONMENTAL CONSIDERATION

The proposed changes to the emergency action levels maintain the environmental bounds of the current environmental assessment associated with the DBNPS. The proposed changes will not affect plant safety and will not have an adverse effect on the probability of an accident occurring. The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure.

Therefore, no environmental impact statement of environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. 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, November 2012," dated March 28, 2013 (ADAMS Accession Number ML12346A463).
2. NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," dated November 2012 (ADAMS Accession Number ML13091A209).
3. Regulatory Issue Summary 2005-02, Revision 1, "Clarifying the Process for Making Emergency Plan Changes," dated August 19, 2011 (ADAMS Accession Number ML100340545).
4. Regulatory Issue Summary 2003-18, Supplement 2, "Use of NEI 99-01, Methodology for Development of Emergency Action Levels," dated December 12, 2005 (ADAMS Accession Number ML051450482).
5. Letter from Eric J. Leeds (NRC) to Barry S. Allen (FENOC), Davis-Besse Nuclear Power Station, Unit No. 1, Safety Evaluation for Emergency Action Levels (TAC No. MD7313), dated December 19, 2008 (ADAMS Accession Number ML083450120).
6. Information Notice 2013-01, "Emergency Action Level Thresholds Outside the Range of Radiation Monitors," dated February 13, 2013 (ADAMS Accession Number ML12325A326).
7. Information Notice 2005-19, "Effect of Plant Configuration Changes on the Emergency Plan", dated July 18, 2005 (ADAMS Accession Number ML051530520).

7.0 ATTACHMENTS

1. Emergency Action Level (EAL) Bases Document
2. Emergency Action Level (EAL) Bases Document (Redline Version)
3. Davis-Besse Nuclear Power Station NEI 99-01, Revision 6, EAL Comparison Matrix
4. Emergency Action Level (EAL) Wallcharts

Evaluation of Proposed License Amendment

Attachment 1

Emergency Action Level (EAL) Bases Document

(272 Pages Follow)

EMERGENCY ACTION LEVEL TECHNICAL BASES DOCUMENT

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 Davis-Besse Nuclear Power Station (DBNPS). It should be used to facilitate review of the DBNPS EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of RA-EP-01500 Emergency Classification, may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Director in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training and for explaining event classifications to off-site officials.

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

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

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 DBNPS Emergency Plan (ref. 4.1.13).

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 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," (ref. 4.1.1), DBNPS 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 (RC): The RC 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 (CT): 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 RC 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 DBNPS EAL scheme includes the following features:

- Division of the EAL set into three broad groups:

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

    - EALs applicable only under hot operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Hot Standby, Startup, or Power Operation mode.

    - EALs applicable only under cold operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling or Defueled mode.

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

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

Recognition category and subcategory titles were selected to represent conditions that are operationally significant to the EAL-user. The DBNPS EAL recognition categories align to and represent the NEI 99-01 Revision 6 “Recognition Categories.” Subcategories are used in the DBNPS scheme to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The DBNPS EAL categories and subcategories are listed below.

EAL Groups, Recognition Categories and Subcategories

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

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

2.5 Technical Bases Information

EAL technical bases are provided in Attachment 1 for each EAL according to EAL group (ANY, Hot, Cold), EAL category (R, C, H, S, F and E) 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 recognition category as described in Section 2.4 (H, R, E, C, S or F)

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

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

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

4. Fourth character (number): The numerical sequence of the EAL within the EAL subcategory. If the subcategory has only one EAL, it is given the number one (1).

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 ALL. (See Section 2.6 for operating mode definitions)

Basis:

A Generic basis section that provides a description of the rationale for the EAL as provided in NEI 99-01 Rev. 6. This is followed by a Plant-Specific basis section that provides DBNPS-relevant information concerning the EAL.

DBNPS Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.6 Operating Mode Applicability (ref. 4.1.8)

Mode	Reactivity Condition (K_{eff})	% Rated Power*	Avg. Reactor Coolant Temperature (ºF)
1) Power Operation	≥ 0.99	> 5%	N/A
2) Startup	≥ 0.99	≤ 5%	N/A
3) Hot Standby	< 0.99	N/A	≥ 280
4) Hot Shutdown	< 0.99	N/A	280 > T_{avg} > 200
5) Cold Shutdown	< 0.99	N/A	≤ 200
6) Refueling	One or more vessel head closure bolts less than fully tensioned.		
D) Defueled	ALL reactor fuel removed from reactor pressure vessel (full core offload during refueling or extended outage).		

Refer to Section 3.3.2 for guidance on event caused mode changes.

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Emergency Director must consider ALL information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes, and the technical basis information. In the Recognition Category F matrices, EALs are based on loss or potential loss of FISSION PRODUCT BARRIER THRESHOLDS.

3.1.1 Classification Timeliness

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, "Interim Staff Guidance, Emergency Planning for Nuclear Power Plants" (ref. 4.1.11).

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.

An indication, report, or condition is considered 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. The validation of indications should be completed in a manner that supports timely emergency declaration.

3.1.3 IMMINENT Conditions

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

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) (ref. 4.1.11).

3.1.6 Emergency Director 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 Emergency Director with the ability to classify events and conditions based upon judgment using EALs that are consistent with the Emergency Classification Level (ECL) definitions (refer to Category H). The Emergency Director will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated in the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

3.1.7 Emergency Action Levels with Embedded Time Requirements

Some EALs have embedded time requirements. Declaration must be made as soon as the Emergency Director recognizes that the conditions will not be successfully resolved within 15 minutes. Therefore, for EALs with time-embedded requirements the time for emergency declaration starts with the initial alarm or indication of the event, not after the embedded time has elapsed.

For EALs with longer embedded time requirements, the 15-minute clock for declaration begins with recognition that the assigned time limit will be exceeded. For example, SG1.1 - “Restoration of at least one essential bus in < 4 hours is **not** likely.” If 20 minutes after loss of the C-1 bus and the D-1 bus it becomes apparent that restoration of either bus will not likely occur within 4 hours the Emergency Director must immediately declare the General Emergency.

3.2 Classification Methodology

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

met, the emergency classification process “clock” starts, and the ECL must be declared in accordance with plant procedures no later than fifteen minutes after the process “clock” started.

When assessing an EAL that specifies a time duration for the off-normal condition, the “clock” for the EAL time duration runs concurrently with the emergency classification process “clock.” For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01 (ref. 4.1.11).

3.2.1 Classification of Multiple Events and Conditions

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

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

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

3.2.3 Classification of IMMINENT Conditions

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

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 be terminated.

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

3.2.5 Classification of Short-Lived Events

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

3.2.6 Classification of Transient Conditions

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

<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.

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

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. 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 ML13091A209

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

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

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

4.1.5 10 § CFR 50.73 License Event Report System

4.1.6 DBNPS ODCM

4.1.7 DBNPS UFSAR Figure 1.2-12 Site Plan

4.1.8 Technical Specifications Table 1.1-1 Modes

4.1.9 DB-OP-06904 Shutdown Operations

4.1.10 NG-QS-00121 DBNPS Writer's Guide

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

4.1.12 Site CSAR NUH-003

4.1.13 DBNPS Emergency Plan

4.1.14 DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tube Rupture

4.2 Implementing

4.2.1 RA-EP-01500 Emergency Classification

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

4.2.3 DBNPS EAL Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

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

Alert

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

Confinement Boundary

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the DBNPS Dry Fuel Storage Facility, CONFINEMENT BOUNDARY is defined as the Dry Shielded Canister (DSC) (ref. 4.1.12).

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 existing plant conditions (ref. 4.1.9).

Emergency Action Level (EAL)

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

Emergency Classification Level (ECL)

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

- Unusual Event (UE)
- Alert
- Site Area Emergency (SAE)
- General Emergency (GE)

EPA PAGs

Environment Protection Agency Protective Action Guidelines. The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs requires DBNPS 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 may require a post-event inspection to determine if the attributes of an EXPLOSION are present.

Faulted

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

Fire

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

Fission Product Barrier Threshold

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

Flooding

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

General Emergency

Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Hostage

A person(s) held as leverage against the station to ensure that demands will be met by the station.

Hostile Action

An act toward DBNPS 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 DBNPS. Non-terrorism-based EALs should be

used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

Hostile Force

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

Imminent

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

Impede(d)

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

Independent Spent Fuel Storage Installation (ISFSI)

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

Initiating Condition (IC)

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

Maintain

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

Owner Controlled Area

The property associated with the station and owned by the company. Access is normally limited to persons entering for official business (ref. 4.1.13).

Projectile

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

Protected Area

An area that normally encompasses all controlled areas within the security protected area fence.

RCS Intact

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

Refueling Pathway

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

Restore

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

Ruptured

The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection (ref. 4.1.14).

Safety System

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

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

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

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

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

Security Condition

ANY Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

Site Area Emergency

Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the SITE BOUNDARY.

Site Boundary

Area as depicted in UFSAR Figure 1.2-12 Site Plan (ref. 4.1.7). The SITE BOUNDARY is defined at a minimum exclusion distance of 0.75 miles. This is the nearest distance from potential release points at which protective actions would be required for members of the public.

Unisolable

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

Unplanned

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

Unusual Event

Events are in progress or have occurred which indicate a potential degradation 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.

Valid

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

Visible Damage

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

5.2 Abbreviations/Acronyms

°F ..... Degrees Fahrenheit
° ..... Degrees
AC ..... Alternating Current
AOP ..... Abnormal Operating Procedure
ATWS ..... Anticipated Transient Without Scram
B&W ..... Babcock & Wilcox
BWST ..... Borated Water Storage Tank
DBNPS ..... Davis-Besse Nuclear Power Station
CCW ..... Component Cooling Water
CDE ..... Committed Dose Equivalent
CFR ..... Code of Federal Regulations
DBA ..... Design Basis Accident
DC ..... Direct Current
DFSF ..... Dry Fuel Storage Facility
DHR ..... Decay Heat Removal
DSC ..... Dry Shielded Canister
EAL ..... Emergency Action Level
ECCS ..... Emergency Core Cooling System
ECL ..... Emergency Classification Level
EOF ..... Emergency Operations Facility
EOP ..... Emergency Operating Procedure
EPA ..... Environmental Protection Agency
EPIP ..... Emergency Plan Implementing Procedure
ERG ..... Emergency Response Guideline
ESF ..... Engineered Safety Feature
FAA ..... Federal Aviation Administration
FBI ..... Federal Bureau of Investigation
FEMA ..... Federal Emergency Management Agency
FSAR ..... Final Safety Analysis Report
GE ..... General Emergency
IC ..... Initiating Condition
IPEEE ..... Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI ..... Independent Spent Fuel Storage Installation

K_{eff} ..... Effective Neutron Multiplication Factor
LCO ..... Limiting Condition of Operation
LER ..... Licensee Event Report
LOCA ..... Loss of Coolant Accident
MSIV ..... Main Steam Isolation Valve
MSL ..... Main Steam Line
MSSV ..... Main Steam Safety Valve
mR, mRem, mrem, mREM ..... milli-Roentgen Equivalent Man
MU-HPI ..... Makeup and High Pressure Injection
MW ..... Megawatt
NEI ..... Nuclear Energy Institute
NESP ..... National Environmental Studies Project
NRC ..... Nuclear Regulatory Commission
NSSS ..... Nuclear Steam Supply System
NORAD ..... North American Aerospace Defense Command
OBE ..... Operating Basis Earthquake
OCA ..... Owner Controlled Area
ODCM ..... Off-site Dose Calculation Manual
ORO ..... Offsite Response Organization
PA ..... Protected Area
PAG ..... Protective Action Guideline
PORV ..... Power Operated Relief Valve
PRA/PSA ..... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR ..... Pressurized Water Reactor
PSIG ..... Pounds per Square Inch Gauge
R ..... Roentgen
RCDT ..... Reactor Coolant Drain Tank
RCS ..... Reactor Coolant System
Rem, rem, REM ..... Roentgen Equivalent Man
RPS ..... Reactor Protection System
RV ..... Reactor Vessel
SAR ..... Safety Analysis Report
SBO ..... Station Blackout
SBODG ..... Station Blackout Diesel Generator

SCBA ..... Self-Contained Breathing Apparatus
SG ..... Steam Generator
SI ..... Safety Injection
SPDS ..... Safety Parameter Display System
SRO ..... Senior Reactor Operator
TEDE ..... Total Effective Dose Equivalent
TOAF ..... Top of Active Fuel
TSC ..... Technical Support Center
TRM ..... Technical Requirements Manual

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

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

DBNPS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
HU1.1	HU1	1
HU1.2	HU1	2
HU1.3	HU1	3
HU2.1	HU2	1
HU3.1	HU3	1
HU3.2	HU3	2
HU3.3	HU3	3
HU3.4	HU3	4
HU4.1	HU4	1
HU4.2	HU4	2
HU4.3	HU4	3
HU4.4	HU4	4
HU7.1	HU7	1
HA1.1	HA1	1
HA1.2	HA1	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

DBNPS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
HG1.1	HG1	1
HG7.1	HG7	1
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
EU1.1	E-HU1	1
CU1.1	CU1	1
CU1.2	CU1	2

DBNPS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
CU2.1	CU2	1
CU3.1	CU3	1
CU3.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1
CU5.2	CU5	2
CU5.3	CU5	3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA3.1	CA3	1, 2
CA6.1	CA6	1
CS1.1	CS1	3
CG1.1	CG1	2
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	1
SU4.2	SU3	2
SU5.1	SU4	1
SU6.1	SU5	1
SU6.2	SU5	2
SU7.1	SU6	1
SU7.2	SU6	2
SU7.3	SU6	3
SU8.1	SU7	1

DBNPS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
SU8.2	SU7	2
SA1.1	SA1	1
SA3.1	SA2	1
SA6.1	SA5	1
SA9.1	SA9	1
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1
SG1.1	SG1	1
SG1.2	SG8	1
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1

7.0 ATTACHMENTS

Attachment 1, Emergency Action Level Technical Bases

Attachment 2, Fission Product Barrier Matrix and Basis

Attachment 3, Safe Shutdown Rooms/Areas Tables R-2 & H-2 Bases

Category H – Hazards and Other Conditions Affecting Plant Safety

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

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

1. Security

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

2. Seismic Event

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

3. Natural or Technology Hazard

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

4. Fire

FIREs can pose significant hazards to personnel and reactor safety. Appropriate for classification are FIREs within the site PROTECTED AREA or FIREs that may affect operability of equipment needed for safe shutdown.

5. Hazardous Gas

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

6. Control Room Evacuation

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

7. Emergency Director Judgment

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

category provide the Emergency Director the latitude to classify emergency conditions consistent with the established classification criteria based upon Emergency Director judgment.

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility

EAL:

HG1.1 General Emergency

A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor

AND EITHER of the following has occurred:

ANY of the following safety functions **cannot** be controlled or maintained

- Reactivity
- Core cooling (RCS inventory)
- RCS heat removal (ability to maintain heat sink)

OR

Damage to spent fuel has occurred or is IMMINENT

Mode Applicability:

ALL

NEI 99-01 Basis:

This IC addresses an event in which a HOSTILE FORCE has taken physical control of the facility to the extent that the plant staff can no longer operate equipment necessary to maintain key safety functions. It also addresses a HOSTILE ACTION leading to a loss of physical control that results in actual or IMMINENT damage to spent fuel due to 1) damage to a spent fuel pool cooling system (e.g., pumps, heat exchangers, controls, etc.) or, 2) loss of spent fuel pool integrity such that sufficient water level cannot be maintained.

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

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 DBNPS Physical Security Plan (ref.1).

DBNPS Basis:

None

DBNPS Basis Reference(s):

1. DBNPS Physical Security Plan (safeguards)
2. DB-OP-02544 Security Events or Threats (restricted)
3. RA-EP-02890 Emergency Response Organization Response to Security Events or Threats
4. NEI 99-01 HG1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA

EAL:

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

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

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

DBNPS Basis:

The Security Shift Supervision is defined as the Security Shift Supervisor or designee.

These individuals 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 is highly controlled due to the strict secrecy controls placed on the DBNPS Physical Security Plan (Safeguards) information (ref. 1).

DBNPS Basis Reference(s):

1. DBNPS Physical Security Plan (safeguards)
2. DB-OP-02544 Security Events or Threats (restricted)
3. RA-EP-02890 Emergency Response Organization Response to Security Events or Threats
4. NEI 99-01 HS1

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HA1.1 Alert

A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor

Mode Applicability:

ALL

NEI 99-01 Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA.

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

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

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

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

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

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 DBNPS Physical Security Plan (ref. 1).

DBNPS Basis:

The Security Shift Supervision is defined as the Security Shift Supervisor or designee.

Escalation of the emergency classification level would be via IC HS1 or HG1.

DBNPS Basis Reference(s):

1. DBNPS Physical Security Plan (safeguards)
2. DB-OP-02544 Security Events or Threats (restricted)
3. RA-EP-02890 Emergency Response Organization Response to Security Events or Threats
4. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HA1.2	**Alert**
A validated notification from NRC of an aircraft attack threat within 30 min. of the site	

Mode Applicability:

ALL

NEI 99-01 Basis:

This IC addresses the notification of an aircraft attack threat. This event will require rapid response and assistance due to the need to prepare the plant and staff for a potential aircraft impact.

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

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

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

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

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

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

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

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

DBNPS Basis:

The Security Shift Supervision is defined as the Security Shift Supervisor or designee.

Escalation of the emergency classification level would be via IC HS1 or HG1.

DBNPS Basis Reference(s):

1. DBNPS Physical Security Plan (safeguards)
2. DB-OP-02544 Security Events or Threats (restricted)
3. RA-EP-02890 Emergency Response Organization Response to Security Events or Threats
4. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1 Unusual Event

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3, 4). Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

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

This EAL references the Security Shift Supervisor because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.39 information.

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 DBNPS Physical Security Plan (ref. 1).

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

DBNPS Basis:

The Security Shift Supervision is defined as the Security Shift Supervisor or designee.

This EAL is based on the DBNPS Physical Security Plan (ref. 1).

DBNPS Basis Reference(s):

1. DBNPS Physical Security Plan (safeguards)
2. DB-OP-02544 Security Events or Threats (restricted)
3. RA-EP-02890 Emergency Response Organization Response to Security Events or Threats
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.2	**Unusual Event**
Notification of a credible security threat directed at the site	

Mode Applicability:

ALL

NEI 99-01 Basis:

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

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3, 4). Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

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

This EAL addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the DBNPS Physical Security Plan (ref. 1).

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 DBNPS Physical Security Plan (ref. 1).

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

DBNPS Basis:

The Security Shift Supervision is defined as the Security Shift Supervisor or designee.

This EAL is based on the DBNPS Physical Security Plan (ref. 1).

DBNPS Basis Reference(s):

1. DBNPS Physical Security Plan (safeguards)
2. DB-OP-02544 Security Events or Threats (restricted)
3. RA-EP-02890 Emergency Response Organization Response to Security Events or Threats
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.3	**Unusual Event**
A validated notification from the NRC providing information of an aircraft threat	

Mode Applicability:

ALL

NEI 99-01 Basis:

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

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3, 4). Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

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

This EAL addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with the DBNPS Physical Security Plan (ref. 1).

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 DBNPS Physical Security Plan (ref. 1).

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

DBNPS Basis:

The Security Shift Supervision is defined as the Security Shift Supervisor or designee.

This EAL is based on the DBNPS Physical Security Plan (ref. 1).

DBNPS Basis Reference(s):

1. DBNPS Physical Security Plan (safeguards)
2. DB-OP-02544 Security Events or Threats (restricted)
3. RA-EP-02890 Emergency Response Organization Response to Security Events or Threats
4. NEI 99-01 HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE levels

EAL:

HU2.1	**Unusual Event**
Seismic event > OBE as indicated by OBE alarm on seismic alarm panel C5764A	

Mode Applicability:

ALL

NEI 99-01 Basis:

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

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event (e.g., lateral accelerations in excess of 0.08g). The Shift Manager or Emergency Director 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.

DBNPS Basis:

Ground motion acceleration of 0.08g horizontal or 0.053g vertical is the Maximum Probable Earthquake as is considered generically as the Operating Basis Earthquake for DBNPS (ref. 1).

Four triaxial strong motion accelerographs are installed within plant structures. A twelve-channel magnetic tape acceleration recording device is used to record all triaxial accelerometer transducer outputs. The recording device is located in the control room with a playback unit featuring a visual display. The time-history recorded at each accelerograph location can be analyzed to determine its corresponding peak acceleration values and to verify that site Operating Basis Earthquake (OBE) limits have not been exceeded. Immediate

control room alarm indication of an earthquake of 0.08 g or greater is annunciated on the seismic control panel (C5764A), following seismic trigger actuation by the Containment Concrete Foundation accelerograph (ref. 2).

RA-EP-02820 Earthquake provides the guidance for determining any required response actions if the OBE earthquake threshold is exceeded (ref. 3).

When the Seismic Monitoring Cabinet key locked switch is placed in OFF, ALL OBE and SSE alarms and cabinet recording functions (TNC 8.3.3 functions 1 and 3 and the DBRM-EMER-5003 Seismic Event Detection function) are nonfunctional. Therefore, the Compensatory Measures listed in DBRM-EMER-5003, Equipment Important to Emergency Response for a loss of the Seismic Event Detection function are required to be employed by the operators to determine if an earthquake occurs and to determine the magnitude of the event in a timely manner" (ref. 5).

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. However, such confirmation should not preclude a timely emergency declaration based on receipt of the OBE alarm. Contact the NEIC by calling **(303) 273-8500**. Select **option #1**, then **#2** and inform the analyst you wish to confirm recent seismic activity near DBNPS. Provide the analyst with the following DBNPS coordinates: **41º 35' 49" north latitude, 83º 05' 16" west longitude** and exact time of seismic activity (ref. 4). Alternatively, access near real-time seismic activity via the NEIC website:

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

DBNPS Basis Reference(s):

1. Updated FSAR Section 3.1 Seismic Design
2. Updated FSAR Section 3.7.4 Criteria for Seismic Instrumentation Program
3. RA-EP-02820 Earthquake
4. UFSAR Section 2.1.1 Site Location
5. CA 2012-04838-08
6. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

DBNPS Basis:

Response actions associated with a tornado onsite is provided in RA-EP-02810 Tornado or High Winds (ref. 1).

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

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

DBNPS Basis Reference(s):

1. RA-EP-02810 Tornado or High Winds
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.2 Unusual Event

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

DBNPS Basis:

Areas containing safe shutdown equipment susceptible to internal FLOODING are Service Water Pump Room and adjacent areas, Component Cooling Water Pump Room and Emergency Core Cooling System Room(s) (ref.1). Refer to EAL CA6.1 or SA9.1 for internal FLOODING affecting one or more SAFETY SYSTEM trains.

DBNPS Basis Reference(s):

1. RA-EP-02880 Internal Flooding
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 PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

DBNPS Basis:

As used here, the term "offsite" means the areas external to the DBNPS PROTECTED AREA.

DBNPS Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.4 Unusual Event

A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

This EAL is not intended 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.

DBNPS Basis:

As used here, the term "onsite" means the areas within the DBNPS OWNER CONTROLLED AREA.

DBNPS Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – FIRE

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

EAL:

HU4.1 Unusual Event

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

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

AND

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

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

Table H-1 Safe Shutdown Fire Areas
• Containment • Control Room • Auxiliary Building • Intake Structure • Borated Water Storage Tank

Mode Applicability:

ALL

NEI 99-01 Basis:

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

For EAL HU4.1 the intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

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

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

DBNPS Basis:

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

Table H-1 Fire Areas are based on DBNPS Unit 1 Fire Hazard Analysis Report. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1, 2).

One of the following 4 things must occur within 15-minutes of the original FIRE indication / notification (ref. 3, 4):

1. The alarm must be proven to be spurious (there is no FIRE),
2. The FIRE must be extinguished,
3. The FIRE must be proven not to be in a Table H-1 area, or
4. The Unusual Event must be declared.

For the purposes of declaring an emergency event, the term "extinguished" means no visible flames.

The 15 minute time period starts with EITHER:

1. A credible notification to the Control Room that a FIRE is occurring in a table H-1 area, (verbal notification from the scene) or
2. Control Room indication of multiple fire detection system alarms/actuations in a Table H-1 area

DBNPS Basis Reference(s):

1. DBNPS Fire Hazard Analysis Report
2. DB-OP-02529 Fire Procedure
3. Condition Report 09-66994, Emergency Action Level Clarification
4. Condition Report 09-69475, White Finding Identified For Inadequate Emergency Classification of Event
5. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – FIRE

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

EAL:

HU4.2 Unusual Event

Receipt of a single fire alarm (i.e., no other indications of a FIRE)

AND

The fire alarm is indicating a FIRE within **ANY** Table H-1 area

AND

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

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

Table H-1 Safe Shutdown Fire Areas
• Containment • Control Room • Auxiliary Building • Intake Structure • Borated Water Storage Tank

Mode Applicability:

ALL

NEI 99-01 Basis:

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

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

Basis-Related Requirements from Appendix R

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

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

DBNPS Basis:

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

Table H-1 Fire Areas are based on DBNPS-1465.00-00-0006 Design Basis Specification for the Plant Fire Protection and AP/0/A/5500/045 Plant Fire. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1, 2).

One of the following 4 things must occur within 30 minutes of the original FIRE indication / notification (ref. 3, 4):

1. The alarm must be proven to be spurious (there is no FIRE),
2. The FIRE must be extinguished,
3. The FIRE must be proven not to be in a Table H-1 area, or
4. The Unusual Event must be declared.

For the purposes of declaring an emergency event, the term "extinguished" means no visible flames.

The 30 minute time period starts with Control Room indication of a fire detection system alarm/actuation (any single alarm or a single actuation of a system).

DBNPS Basis Reference(s):

1. DBNPS Fire Hazard Analysis Report
2. DB-OP-02529 Fire Procedure
3. Condition Report 09-66994, Emergency Action Level Clarification
4. Condition Report 09-69475, White Finding Identified For Inadequate Emergency Classification of Event
5. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – FIRE

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

EAL:

HU4.3 Unusual Event

A FIRE within the plant PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

In addition to a FIRE addressed by EAL HU4.1 or HU4.2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety.

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

DBNPS Basis:

None

DBNPS 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 plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

DBNPS Basis:

None

DBNPS 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 Safe Shutdown 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 Shutdown Rooms/Areas

Room/Area	Mode Applicability
Aux Bldg. 565' ele. Room 236 #2 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 585' ele. Room 304 corridor outside #3 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 603' ele. Room 427 - #2 Electrical Penetration Room	1, 2, 3

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 – Hot Standby

NEI 99-01 Basis:

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

An Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.

Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the Emergency Director judgment that the gas concentration in the affected room/area is sufficient

to preclude or significantly IMPEDE procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as IMPEDED if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

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

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures, which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or IMPEDE a required action.
- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death. A steam leak of significant magnitude that prevents access to the area should be considered an asphyxiant.

This EAL does not apply to firefighting activities that automatically or manually activate a fire suppression system in an area.

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

DBNPS Basis:

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

The list of plant rooms or areas with entry-related mode applicability identified specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or

areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

DBNPS Basis Reference(s):

1. Attachment 3 Safe Shutdown Rooms/Areas Tables R-2 & H-2 Bases
2. NEI 99-01 HA5

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

Initiating Condition: Inability to control a key safety function from outside the Control Room

EAL:

HS6.1 Site Area Emergency

An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel

AND

Control of **ANY** of the following key safety functions is not reestablished within 15 min. (Note 1):

- Reactivity
- Core Cooling (RCS inventory)
- RCS heat removal (ability to maintain heat sink)

Note 1: The Emergency Director 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

NEI 99-01 Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not “control” is established at the remote safe shutdown location(s) is based on Emergency Director judgment. The Emergency Director is expected to make a reasonable, informed judgment within 15-minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

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

DBNPS Basis:

Shift Manager determines the Control Room requires evacuation. Control Room may become uninhabitable by FIRE, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (ref. 1, 2).

The 15-minute time for transfer is based on analysis or assessments as to how quickly control must be reestablished without core uncovering and/or core damage. The 15-minute time period starts when EITHER 1) control of the plant is no longer maintained in the Control Room or 2) the last operator has left the Control Room, whichever comes first.

DBNPS Basis Reference(s):

1. DB-OP-02508 Control Room Evacuation
2. DB-OP 02519 Serious Control Room Fire
3. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

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

EAL:

HA6.1 Alert

An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel

Mode Applicability:

ALL

NEI 99-01 Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

This EAL is only applicable when the decision has been made to evacuate the Control Room, and not when conditions that may require evacuation are being evaluated by referring to either DB-OP-02519 or DB-OP-02508 (ref. 1, 2).

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

DBNPS Basis:

Shift Manager (SM) determines Control Room requires evacuation. Control Room inhabitability may be caused by FIRE, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (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.

DBNPS Basis Reference(s):

1. DB-OP-02508 Control Room Evacuation
2. DB-OP 02519 Serious Control Room Fire
3. NEI 99-01 HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency

EAL:

HG7.1	**General Emergency**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	

Mode Applicability:

ALL

NEI 99-01 Basis:

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

DBNPS Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the DBNPS Emergency Response Plan. The Operations Shift Manager(SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, 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 (ref. 1).

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

DBNPS Basis Reference(s):

1. DBNPS Emergency Plan section 5.2 DBNPS Emergency Management
2. NEI 99-01 HG7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency

EAL:

HS7.1 Site Area Emergency

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. **ANY** releases are **not** expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the SITE BOUNDARY.

Mode Applicability:

ALL

NEI 99-01 Basis:

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

DBNPS Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the DBNPS Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, 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 (ref. 1).

DBNPS Basis Reference(s):

1. DBNPS Emergency Plan section 5.2 DBNPS Emergency Management
2. NEI 99-01 HS7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

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

EAL:

HA7.1 Alert

Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. **ANY** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Mode Applicability:

ALL

NEI 99-01 Basis:

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

DBNPS Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the DBNPS Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, 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 (ref.1).

DBNPS Basis Reference(s):

1. DBNPS Emergency Plan section 5.2 DBNPS Emergency Management
2. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a UE

EAL:

HU7.1 Unusual Event

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Mode Applicability:

ALL

NEI 99-01 Basis:

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

DBNPS Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the DBNPS Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, 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 (ref. 1).

DBNPS Basis Reference(s):

1. DBNPS Emergency Plan section 5.2 DBNPS Emergency Management
2. NEI 99-01 HU7

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 radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

EAL:

RG1.1	**General Emergency**
Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E +00 µCi/cc for 15 minutes or longer (Notes 1, 2, 3, 4)	

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

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

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

DBNPS Basis:

The DBNPS gaseous effluent limits for RG1.1 are based on values that equate to an offsite dose greater than EITHER:

- 1000 mrem TEDE
- 5000 mrem CDE Thyroid

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

The station vent is the primary effluent monitor release point. Other pathways such as Safety and relief valves, as well as AFW and MSSV, do not have direct effluent release monitors. Releases from these points would be verified and monitored by RG1.3 survey readings.

The “GE” Station Vent gaseous effluent monitor value corresponds to calculated doses of 100% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) per EP-EALCALC-DB-0703 Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1 and RU1) (ref. 1).

DBNPS Basis Reference(s):

1. EP-EALCALC-DB-0703 Rev. 2 Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1 and RU1)
2. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

DBNPS Basis:

Dose assessments are performed by computer-based methods (ref. 1, 2).

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

DBNPS Basis Reference(s):

1. NOP-LP-5402 Davis-Besse MIDAS Dose Assessment Software
2. RA-EP-02240 Offsite Dose Assessment
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.3 General Emergency

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

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

(Notes 1, 2)

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

DBNPS Basis:

NOP-LP-5015 FENOC Field Monitoring Teams Radiation Monitoring Teams Field Surveys provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

DBNPS Basis Reference(s):

1. NOP-LP-5015 FENOC Field Monitoring Teams Radiation Monitoring Teams Field Surveys
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 100 mrem TEDE or 500 mrem thyroid CDE

EAL:

RS1.1 Site Area Emergency

Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E -01 µCi/cc for 15 minutes or longer (Notes 1, 2, 3, 4)

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

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

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

DBNPS Basis:

The DBNPS gaseous effluent limits for RS1.1 are based on values that equate to an offsite dose greater than EITHER:

- 100 mrem TEDE
- 500 mrem CDE Thyroid

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

The station vent is the primary effluent monitor release point. Other pathways such as Safety and relief valves, as well as AFW and MSSV, do not have direct effluent release monitors. Releases from these points would be verified and monitored by RS1.3 survey readings.

The “SAE” Station Vent gaseous effluent monitor value corresponds to calculated doses of 10% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) per EP-EALCALC-DB-0703 Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1 and RU1) (ref. 1).

DBNPS Basis Reference(s):

1. EP-EALCALC-DB-0703 Rev.2 Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1 and RU1)
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

DBNPS Basis:

Dose assessments are performed by computer-based or manual methods (ref. 1, 2).

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

DBNPS Basis Reference(s):

1. NOP-LP-5402 Davis-Besse MIDAS Dose Assessment Software
2. RA-EP-02240 Offsite Dose Assessment
3. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.3 Site Area Emergency

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

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

(Notes 1, 2)

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

DBNPS Basis:

NOP-LP-5015 FENOC Field Monitoring Teams Radiation Monitoring Teams Field Surveys provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

DBNPS Basis Reference(s):

1. NOP-LP-5015 FENOC Field Monitoring Teams Radiation Monitoring Teams Field Surveys
2. NEI 99-01 AS1

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

Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E -02 µCi/cc for 15 minutes or longer (Notes 1, 2, 3, 4)

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

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

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

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

DBNPS Basis:

The DBNPS gaseous effluent limits for RA1.1 are based on values that equate to an offsite dose greater than EITHER:

- 10 mrem TEDE
- 50 mrem CDE Thyroid

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

The “Alert” Station Vent gaseous effluent monitor value corresponds to calculated doses of 1% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) per EP-EALCALC-DB-0703 Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1 and RU1) (ref. 1).

The station vent is the primary effluent monitor release point. Other pathways such as Safety and relief valves, as well as AFW and MSSV, do not have direct effluent release monitors. Releases from these points would be verified and monitored by RA1.3 survey readings.

DBNPS Basis Reference(s):

1. EP-EALCALC-DB-0703 Rev.2 Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1 and RU1)
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.2 Alert

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

DBNPS Basis:

Dose assessments are performed by computer-based and manual methods (ref. 1, 2).

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

DBNPS Basis Reference(s):

1. NOP-LP-5402 Davis-Besse MIDAS Dose Assessment Software
2. RA-EP-02240 Offsite Dose Assessment
3. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.3 Alert

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

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

DBNPS Basis:

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

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

DBNPS Basis Reference(s):

1. DBNPS ODCM
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.4 Alert

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

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

(Notes 1, 2)

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

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

DBNPS Basis:

NOP-LP-5015 FENOC Field Monitoring Teams Radiation Monitoring Teams Field Surveys provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

DBNPS Basis Reference(s):

1. NOP-LP-5015 FENOC Field Monitoring Teams Radiation Monitoring Teams Field Surveys
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.1 Unusual Event

Station Vent Channel 1 Noble Gas (RE 4598AA/BA) > 5.72E-03 µCi/cc for 60 minutes or longer

OR

ANY of the following effluent monitors > 2 times the high alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer:

- Waste Gas System Outlet (RE 1822A or B).
- Clean Waste System Outlet (RE 1770A or B).
- Miscellaneous Waste System Outlet (RE 1878A or B).
- Discharge permit specified monitor.

(Notes 1, 2, 3)

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

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

This IC addresses a potential decrease in the level of safety of the plant as indicated by a low-level radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release). It includes ANY gaseous or liquid radiological release,

monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.

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

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

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

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

This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways or 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).

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

DBNPS Basis:

The “UE” gaseous and liquid release values represent two times the appropriate ODCM release rate limits associated with the specified monitors (ref. 2).

<u>Gaseous Releases</u>

Instrumentation that may be used to assess this EAL is listed below (ref. 1):

- Station Vent – RE 4598AA or BA
- Waste Gas System Outlet – RE 1822 A or B (batch release)

The Station Vent gaseous effluent (RE 4598AA/BA) value of 2 times the ODCM setpoint was determined using formulas, isotopic dose conversion factors and meteorology data as specified in the ODCM based on a normal operating isotopic mixture (no clad damage condition). Assumptions and calculation inputs are provided in EP-EALCALC-DB-0703 Rev. 2 (ref. 2).

The Waste Gas Outlet (RE 1822 A or B for batch release) are the release monitors normally used for planned discharges. If a discharge is performed using a different flow path or effluent monitor (e.g., a portable or temporary effluent monitor), then the declaration criteria will be based on the monitor specified in the discharge permit.

Liquid Releases

Instrumentation that may be used to assess this EAL is listed below (ref. 1):

- Clean Waste System Outlet – RE 1770 A or B (batch release)
- Misc. Waste System Outlet – RE 1878 A or B (batch release)

The liquid release monitors listed are those normally used for planned discharges. If a discharge is performed using a different flow path or effluent monitor (e.g., a portable or temporary effluent monitor), then the declaration criteria will be based on the monitor specified in the discharge permit.

DBNPS Basis Reference(s):

1. DBNPS ODCM
2. EP-EALCALC-DB-0703 Rev. 2 Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1 and RU1)
3. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.2 Unusual Event

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

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

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

DBNPS Basis:

Grab samples are used to determine release concentrations or release rates, confirm meter readings, or indicate the need for sampling when the effluent monitors are not in service or other alarms occur. The maximum instantaneous release rate limits are calculated in accordance with the ODCM. These are indicated on approved discharge permit release packages (ref. 1).

DBNPS Basis Reference(s):

1. DBNPS ODCM
2. NEI 99-01 AU1

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. for ≥ 60 min. (Note 1)	

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

DBNPS Basis:

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

Indicated level of 1 ft. on LI4801A (primary) or LI4801B (Backup) corresponds approximately to 1 ft. above the top of the SFP racks (ref. 2).

DBNPS Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. Engineering Change Package 13-0596
3. NEI 99-01 AG2

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.	

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

DBNPS Basis:

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

Indicated level of 1 ft. on LI4801A (primary) or LI4801B (Backup) corresponds approximately to 1 ft. above the top of the SFP racks (ref. 2).

DBNPS Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. Engineering Change Package 13-0596
3. NEI 99-01 AS2

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

NEI 99-01 Basis:

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

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

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

<u>EAL #1</u>

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

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

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

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

DBNPS Basis:

None

DBNPS Basis Reference(s):

1. DB-OP-02003 3-1-B, SFP LVL
2. DB-OP-02003 3-1-A, REFUELING CANAL LVL
3. DB-OP-02547 Spent Fuel Pool Cooling Malfunctions
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:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity as indicated by a high radiation alarm on **ANY** of the following radiation monitor indications:

- RE 8426 SFP Area
- RE 8427 SFP Area
- RE 8417 Fuel Handling Area
- RE 8418 Fuel Handling Area
- RE 8425 Equipment Hatch Area
- RE 8446/8447 Fuel Handling Exhaust
- RE 4598AA/BA Station Vent

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

DBNPS Basis:

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

The high alarm setpoints for the radiation monitors are set to be indicative of significant increases in area and/or airborne radiation (ref. 2).

DBNPS Basis Reference(s):

1. DB-OP-02530 Fuel Handling Accident
2. Radiation Setpoints Manual
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:

RA2.3 **Alert**
Lowering of spent fuel pool level to 10 ft.

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

Escalation of the emergency classification level would be via IC RS1or RS2.

DBNPS Basis:

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

Indicated level of 10 ft. on LI4801A (primary) or LI4801B (Backup) corresponds to plant elevation 588 ft. 6 in. (ref. 2).

DBNPS Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. Engineering Change Package 13-0596
3. NEI 99-01 AA2

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 **ANY** of the following:

- SFP level alarm or indication (LI 1600)
- SFP level indication (LI 4801 A/B)
- Refueling Canal low water level alarm or indication (LI 1627)
- Visual observation

AND

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

- RE 8426 SFP Area
- RE 8427 SFP Area
- RE 8417 Fuel Handling Area
- RE 8418 Fuel Handling Area
- RE 8425 Equipment Hatch Area

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

DBNPS Basis:

The SFP low water level alarm setpoint is actuated by LS 1600 at a setpoint of 23.2 feet on LI 1600 (ref. 1).

The refueling canal low water level alarm setpoint is actuated by LS 1627 at a setpoint of 23.4 feet on LI 1627 (this corresponds to 1.5 inches below normal) (ref. 2).

The LI 4801 A/B is indication only, with no alarm or computer point function.

Water level restoration instructions are performed in accordance with AOPs (ref. 1, 2, 3).

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

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

DBNPS Basis Reference(s):

1. DB-OP-02003 3-1-B, SFP LVL
2. DB-OP-02003 3-1-A, REFUELING CANAL LVL
3. DB-OP-02547 Spent Fuel Pool Cooling Malfunctions
4. NEI 99-01 AU2

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 (RE 8430 or 8431)
- Central Alarm Station (RE 8435 or 8436)

Mode Applicability:

ALL

NEI 99-01 Basis:

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or IMPEDE personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Emergency Director should consider the cause of the increased radiation levels and determine if another IC may be applicable.

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

DBNPS Basis:

Areas that meet this threshold include the Control Room and the Central Alarm Station (CAS). RE 8430 and 8431 monitors the Control room for area radiation. The CAS area does not have installed radiation monitoring but RE 8435 and 8436 are in near proximity and can be used to approximate CAS area radiation levels (ref. 1, 2). The CAS is included because of its importance to permitting access to areas required to assure safe plant operations.

DBNPS Basis Reference(s):

1. DB-OP-06412 Process and Area Radiation Monitor
2. SD-017B System Description for Area Radiation Monitors
3. 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 Safe Shutdown 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 Shutdown Rooms/Areas

Room/Area	Mode Applicability
Aux Bldg. 565' ele. Room 236 #2 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 585' ele. Room 304 corridor outside #3 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 603' ele. Room 427 - #2 Electrical Penetration Room	1, 2, 3

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 – Hot Standby

NEI 99-01 Basis:

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or IMPEDE personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Emergency Director should consider the cause of the increased radiation levels and determine if another IC may be applicable.

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

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.

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.

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

DBNPS Basis:

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

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

DBNPS Basis Reference(s):

1. Attachment 3 Safe Shutdown Rooms/Areas Tables R-2 & H-2 Bases
2. RA-EP-02861 Radiological Incidents
3. NEI 99-01 AA3

Category E – <u>Dry Fuel Storage Facility (DFSF)</u>

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

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

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

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

Category: E - DFSF

Sub-category: None

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Notification of Unusual Event

Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel HSM cask > **EITHER** of the following:

- 100 mrem/hr (neutron + gamma) on the HSM cask wall or roof
- 100 mrem/hr (neutron + gamma) on the center of the HSM cask door

Mode Applicability:

ALL

NEI 99-01 Basis:

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

The existence of “damage” is determined by radiological survey. The technical specification multiple of “2 times”, which is also used in Recognition Category R IC RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the “on-contact” dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

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

DBNPS Basis:

The DBNPS DFSF utilizes one design for dry spent fuel storage: the Nuclear Horizontal Modular Storage (NUHOMS) system.

The system consists of Horizontal Storage Modules (HSM) that contains dry shielded canisters (DSCs). The DSC serves as the CONFINEMENT BOUNDARY. The DSC is welded and designed to provide confinement of all radionuclides under normal, off normal, and accident conditions.

CONFINEMENT BOUNDARY is defined as the barrier(s) between areas containing radioactive substances and the environment. Therefore, damage to a CONFINEMENT BOUNDARY must be a confirmed physical breach between the spent fuel and the environment for the DSC.

The values shown represent 2 times the limits specified in the DFSF Site CSAR for radiation external to a loaded DSC (ref. 1).

DBNPS Basis Reference(s):

1. Site CSAR NUH-003 Table 7.3-2 Shielding Analysis Results for 5 year Cooled Fuel NUHOMS-24P System
3. NEI 99-01 E-HU1

Category C – Cold Shutdown / Refueling System Malfunction

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

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

Loss of essential 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 4160 VAC essential buses.

3. RCS Temperature

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

4. Loss of Essential 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 125 VDC buses.

5. Loss of Communications

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

6. Hazardous Event Affecting SAFETY SYSTEMS

Certain hazardous natural and technological events may result in visible damage to or degraded performance of SAFETY SYSTEMS warranting classification.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CG1.1 General Emergency

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

AND

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

- UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT levels of sufficient magnitude to indicate core uncovery
- Containment Radiation Monitor (RE 4596A or B) reading > 16 R/hr
- Refueling Bridge Portable Area Radiation Monitor reading > 30 R/hr
- Erratic Source Range Monitor indication

AND

ANY Containment Challenge indication, Table C-1

Note 1: The Emergency Director 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 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

NEI 99-01 Basis:

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

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

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

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

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

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

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

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

DBNPS Basis:

The lowest measurable RCS level is the elevation of the RCS hot leg mid-loop (0.4 ft. on LI 10596). Therefore, RCS inventory loss relative to the RCS level elevation corresponding to the top of active fuel must be detected by indirect leakage indications (ref. 1, 2, 3). Sump level increases must be evaluated against other potential sources of 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 in areas outside the containment that cannot be isolated, or a sump sample verified by Chemistry, could also be indicative of a loss of RCS inventory (ref. 1).

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed and portable area radiation monitors. RE 4596A or B, Containment Radiation Monitors are located in the containment, a reading greater than 16 R/hr is indicative of core uncovery. The Refuel Bridge Portable Radiation Monitor, when installed during refueling operations, is designed to provide monitoring of radiation due to a fuel handling event or loss of shielding during refueling operations. If this radiation monitor reaches and exceeds 30 R/hr, a loss of inventory indicative of core uncovery has occurred (ref. 2, 3).

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

Three conditions are associated with a challenge to containment integrity:

- CONTAINMENT CLOSURE is not established (Ref. 4).
- In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gases in the containment. However, containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. An explosive mixture can be formed when hydrogen gas concentration in the containment atmosphere is greater than 4% by volume in the presence of oxygen. Hydrogen monitors, although available at all times, are not in service during normal operations. They are started per DB-OP-02000 Table 3. Action is required if the measured concentration reaches 0.6% (notify TSC) and 3% (notify TSC) (ref 5).
- ANY UNPLANNED increase in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of CONTAINMENT CLOSURE capability. UNPLANNED containment pressure increases indicates CONTAINMENT CLOSURE cannot be assured and the containment cannot be relied upon as a barrier to fission product release.

DBNPS Basis Reference(s):

1. DB-OP-02527 Loss of Decay Heat Removal

2. EP-EALCALC-DB-0704 Radiation Monitor Readings for Core Uncovery During Refueling
3. DB-HP-01152 Performance of High Exposure Work
4. DB-OP-06904, Shutdown Operations
5. DB-OP-02000 Table 3, Containment Monitoring and Control
6. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.1 Site Area Emergency

RCS level cannot be monitored for ≥ 30 min. (Note 1)
AND
Core uncovery is indicated by **ANY** of the following:

- UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT levels of sufficient magnitude to indicate core uncovery
- Containment Radiation Monitor (RE 4596A or B) reading > 16 R/hr
- Refueling Bridge Portable Area Radiation Monitor reading > 30 R/hr
- Erratic Source Range Monitor indication

Note 1: The Emergency Director 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

NEI 99-01 Basis:

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

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

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

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

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

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

DBNPS Basis:

The lowest measurable RCS level is the elevation of the RCS hot leg mid-loop (0.4 ft. on LI 10596). Therefore, RCS inventory loss relative to the RCS level elevation corresponding to the top of active fuel must be detected by indirect leakage indications (ref. 1, 2, 3). Sump level increases must be evaluated against other potential sources of 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 in areas outside the containment that cannot be isolated, or a sump sample verified by Chemistry, could also be indicative of a loss of RCS inventory (ref. 1).

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed and portable area radiation monitors. RE 4596A or B, Containment Radiation Monitors are located in the containment, a reading greater than 16 R/hr is indicative of core uncovery. The Refuel Bridge Portable Radiation Monitor, when installed during refueling operations, is designed to provide monitoring of radiation due to a fuel handling event or loss of shielding during refueling operations. If this radiation monitor reaches and exceeds 30 R/hr, a loss of inventory indicative of core uncovery has occurred (ref. 2, 3).

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

DBNPS Basis Reference(s):

1. DB-OP-02527 Loss of Decay Heat Removal
2. EP-EALCALC-DB-0704 Radiation Monitor Readings for Core Uncovery During Refueling
3. DB-HP-01152 Performance of High Exposure Work
4. NEI 99-01 CS1

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 RCS level ≤ 0.4 ft. (LI 10596)

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

NEI 99-01 Basis:

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

For this EAL, a lowering of RCS water level at or below 0.4 ft. indicates that operator actions have not been successful in restoring and maintaining RCS water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

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

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

DBNPS Basis:

RCS level cannot be measured below the 571 feet elevation (0.4 feet on LI 10596) which is centerline of the hot leg inlet. Should RCS level drop below this point it is assumed water level cannot be monitored other than visually. Continued loss of RCS inventory may result in DHR pump cavitation (ref. 1).

DBNPS Basis Reference(s):

1. DB-OP-02527 Loss of Decay Heat Removal
2. 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 level cannot be monitored for ≥ 15 min. (Note 1)

AND EITHER

- UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

Note 1: The Emergency Director 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

NEI 99-01 Basis:

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

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

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

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

DBNPS Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available. In the Refuel mode, the RCS is not intact and RCS level may be monitored by different means, including the ability to monitor level visually.

RCS level in the Refueling mode is normally monitored using the following instruments:

- LI 10596
- LI10577 A and B
- Clear tubing used for manometer level indication at the RCS cold legs.

RCS level indications (LI 10596, LI 10577A and B and cold leg tubing) provide accurate indication of water level when the RCS is at atmospheric pressure. The reactor vessel flange is at 577 ft. 7 in. (80 in. indicated) (ref. 1).

In this EAL, all RCS water level indication would be unavailable for greater than 15-minutes, and the RCS inventory loss must be detected by indirect leakage indications (ref. 1). Sump and tank level increases must be evaluated against other potential sources of 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, or a sump sample verified by Chemistry, could also be indicative of a loss of RCS inventory.

DBNPS Basis Reference(s):

1. DB-OP-06904 Shutdown Operations
2. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory for 15 minutes or longer

EAL:

CU1.1 Unusual Event

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

Note 1: The Emergency Director 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

NEI 99-01 Basis:

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

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

This EAL recognizes that the minimum required RCS level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.

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

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

DBNPS Basis:

With the plant in Cold Shutdown, RCS water level is normally maintained above a lower level band limit (ref. 1, 2), typically above the pressurizer low level setpoint. However, if RCS level is being controlled below the pressurizer low level setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

With the plant in Refueling mode, RCS water level is normally maintained at or above the reactor vessel flange, (Technical Specification LCO 3.9.6 requires at least 23 ft. of water above the top of the reactor vessel flange in the refueling canal when irradiated fuel is moved in containment). The reactor vessel flange is at 577 ft. 7 in. (80 in. indicated) (ref. 3, 4).

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available. RCS level in the Refueling mode is normally monitored using the following instruments:

- LI 10596
- LI 10577 A and B
- Clear tubing used for manometer level indication at the RCS cold legs.

RCS level indications (LI 10596, LI 10577A and B and cold leg tubing) provide accurate indication of water level when the RCS is at atmospheric pressure (ref. 3).

DBNPS Basis Reference(s):

1. DB-OP-02513 Pressurizer System Abnormal Operation
2. DB-OP-02004 4-2-E PZR LVL LO
3. DB-OP-06904 Shutdown Operations
4. DBNPS Technical Specifications Section 3.9.6 Refueling Canal Water Level
5. DB-OP-02527 Loss of Decay Heat Removal
6. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory for 15 minutes or longer

EAL:

CU1.2 Unusual Event

RCS level **cannot** be monitored

AND EITHER

- UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

NEI 99-01 Basis:

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

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

This EAL addresses a condition where all means to determine 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. 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.

DBNPS Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available. RCS level in the Refueling mode is normally monitored using the following instruments:

- LI 10596
- LI10577 A and B
- Clear tubing used for manometer level indication at the RCS cold legs.

RCS level indications (LI 10596, LI 10577A and B and cold leg tubing) provide accurate indication of water level when the RCS is at atmospheric pressure. The reactor vessel flange is at 577 ft. 7 in. (80 in. indicated) (ref. 1).

In this EAL, all water level indication is unavailable and the RCS inventory loss must be detected by indirect leakage indications (ref. 1). Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. 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, or a sump sample verified by Chemistry, could also be indicative of a loss of RCS inventory.

DBNPS Basis Reference(s):

1. DB-OP-06904 Shutdown Operations
2. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Essential AC Power

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

EAL:

CA2.1 Alert

Loss of **ALL** offsite and **ALL** onsite AC power capability, Table C-2, to essential 4160V buses C1 and D1 for ≥ 15 min. (Note 1)

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

Table C-2 Offsite/Onsite AC Power Sources
Offsite: • X01 • X02 • X11 (back-fed via Main Transformer) **Onsite:** • EDG1 • EDG2 • SBODG

Mode Applicability:

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

NEI 99-01 Basis:

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

When in the Cold Shutdown, Refueling, or Defueled mode, this condition is not classified as a Site Area Emergency because of the increased time available to restore an essential bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this

condition represents an actual or potential substantial degradation of the level of safety of the plant.

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

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

DBNPS Basis:

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses C1 and D1 (ref. 1).

The essential buses during plant operation are normally powered from the 13.8KV offsite power system through their respective 13.8KV/4160V bus tie transformers, via the Unit Auxiliary Transformer (X11). In cold operating modes, the essential buses may be back-fed via the X11 and Main Transformer provided the main generator lead disconnect links are removed (ref. 1, 2). Credit for the X11 back-feed can only be taken if already aligned, as it takes greater than 15 minutes to align.

A standby source of offsite power to each 4160V essential bus is provided from the 13.8KV offsite power system via two separate and independent 13.8KV/4160V Startup Transformers (X01 & X02). Normally each startup transformer serves as a reserve power source for only one essential bus. However, a single startup transformer can be aligned to power both essential buses (ref. 1, 2).

Each essential bus has a diesel generator (EDG1 & EDG2) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power (ref. 1).

An Alternate AC power source, the SBO Diesel Generator, is located onsite. This onsite AC power source can be started from the Control Room and be loaded within 10 minutes of a SBO (ref. 1).

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

DBNPS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. DB-OP-02521 Loss of AC Bus Power Sources
3. NEI 99-01 CA2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Essential AC Power

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

EAL:

CU2.1 Unusual Event

AC power capability, Table C-2, to essential 4160V buses C1 and D1 reduced to a single power source for ≥ 15 min. (Note 1)

AND

ANY additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS

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

Table C-2 Offsite/Onsite AC Power Sources
Offsite: • X01 • X02 • X11 (back-fed via Main Transformer) **Onsite:** • EDG1 • EDG2 • SBODG

Mode Applicability:

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

NEI 99-01 Basis:

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

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

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

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

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

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

DBNPS Basis:

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses C1 and D1 (ref. 1).

The essential buses during plant operation are normally powered from the 13.8KV offsite power system through their respective 13.8KV/4160V bus tie transformers, via the Unit Auxiliary Transformer (X11). In cold operating modes, the essential buses may be back-fed via the X11 and Main Transformer provided the main generator lead disconnect links are removed (ref. 1, 2). Credit for the X11 back-feed can only be taken if already aligned, as it takes greater than 15 minutes to align.

A standby source of offsite power to each 4160V essential bus is provided from the 13.8KV offsite power system via two separate and independent 13.8KV/4160V Startup Transformers (X01 & X02). Normally each startup transformer serves as a reserve power source for only one essential bus. However, a single startup transformer can be aligned to power both essential buses (ref. 1, 2).

Each essential bus has a diesel generator (EDG1 & EDG2) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power (ref. 1).

An Alternate AC power source, the SBO Diesel Generator, is located onsite. This onsite AC power source can be started from the Control Room and be loaded within 10 minutes of a SBO (ref. 1).

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

DBNPS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. DB-OP-02521 Loss of AC Bus Power Sources
3. NEI 99-01 CU2

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-3 duration (Note 1, 10)

OR

UNPLANNED RCS pressure increase > 10 psig due to a loss of RCS cooling (This EAL does not apply during water-solid plant conditions).

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

Note: 10 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.

Table C-3: RCS Heat-up Duration Thresholds

RCS Status	Containment Closure Status	Heat-up Duration
Intact (but **not** reduced inventory)	N/A	60 min.*
Not intact **OR** At reduced inventory	established	20 min.*
	not established	0 min.

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

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

NEI 99-01 Basis:

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

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

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

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

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

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

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

DBNPS Basis:

In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the RCS pressure 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.

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

- Selected Incore thermocouples
- TIRC4B2 Reactor Coolant T-Cold Wide Range (Loop 1)
- TIRC4A2 Reactor Coolant T-Cold Wide Range (Loop 2)
- TIRC3B5 or TIRCB6 Reactor Coolant T-Hot (Loop 1)
- TIRC3A5 or TIRCA6 Reactor Coolant T-Hot (Loop 2)
- DHR Display on SPDS

(ref. 2)

A 10 psig RCS pressure increase can be read on various instruments such as (ref. 2):

- PI 6365B, Loop 1 Pressure

- PI 6365A, Loop 2 Pressure
- PRS RC2A1, Loop 2 Wide Range Pressure
- PI RC2A6, Low Range Pressure

The RCS is considered in reduced inventory if (ref. 3):

- The Reactor Coolant System is not full (loops not filled and the RCS is incapable of natural circulation), OR
- Both steam generators are not available as a heat sink, OR
- The refueling canal level is less than a stable level specified in the Shutdown Defense in Depth Review, typically 23 feet, with the head removed.

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications Table 1.1-1
2. DB-OP-02527 Loss of Decay Heat Removal
3. NG-DB-00117 Shutdown Defense In Depth Assessment
4. NEI 99-01 CA3

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 due to loss of decay heat removal capability	

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

NEI 99-01 Basis:

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 Emergency Director should also refer to IC CA3.

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

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

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

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

DBNPS Basis:

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

- Selected Incore thermocouples

- TIRC4B2 Reactor Coolant T-Cold Wide Range (Loop 1)
- TIRC4A2 Reactor Coolant T-Cold Wide Range (Loop 2)
- TIRC3B5 or TIRCB6 Reactor Coolant T-Hot (Loop 1)
- TIRC3A5 or TIRCA6 Reactor Coolant T-Hot (Loop 2)
- DHR Display on SPDS

(ref. 2)

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.

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications Table 1.1-1 Modes
2. DB-OP-02527 Loss of Decay Heat Removal
3. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

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

Note 1: The Emergency Director 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

NEI 99-01 Basis:

This EAL addresses the inability to determine RCS temperature and level, and represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Director should also refer to IC CA3.

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

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

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

DBNPS Basis:

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

- Selected Incore thermocouples
- TIRC4B2 Reactor Coolant T-Cold Wide Range (Loop 1)
- TIRC4A2 Reactor Coolant T-Cold Wide Range (Loop 2)
- TIRC3B5 or TIRCB6 Reactor Coolant T-Hot (Loop 1)

- TIRC3A5 or TIRCA6 Reactor Coolant T-Hot (Loop 2)
- DHR Display on SPDS

(ref. 2)

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 RCS level may be monitored by different means, including the ability to monitor level visually.

RCS level in the Refueling mode is normally monitored using the following instruments:

- LI 10596
- LI10577 A and B
- Clear tubing used for manometer level indication at the RCS cold legs.

RCS level indications (LI 10596, LI 10577A and B and cold leg tubing) provide accurate indication of water level when the RCS is at atmospheric pressure. However, there is no redundant means of RCS level indication when Mode 6 is entered (ref. 2).

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications Table 1.1-1
2. DB-OP-02527 Loss of Decay Heat Removal
3. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – Loss of Essential DC Power

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

EAL:

CU4.1	**Unusual Event**
< 105 VDC voltage indications on Technical Specification **required** essential 125 VDC distribution panels for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director 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

NEI 99-01 Basis

This IC addresses a loss of essential 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 an essential DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, “required” means the essential 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 essential DC power affecting Train B would require the declaration of an Unusual Event. A loss of essential 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.

DBNPS Basis:

The DC electrical distribution subsystem consists of two trains, designated Train 1 and Train 2. Each train consists of a 250/125 VDC motor control center (MCC), and each 250/125 VDC MCC consists of two 125 VDC buses (one positive and one negative) that can be powered from a polarity specific battery or battery charger. The two DC buses then supply a specific Essential DC Distribution Panel (D1P and D1N for DCMCC 1, and D2P and D2N for DCMCC2). The four Essential DC Distribution Panels independently supply DC electrical

power to the four inverters, which in turn power the 120 VAC essential buses (ref. 1, 2, 3, 4, 5, 6).

The Class 1E DC loads have an operating voltage range of 105 to 135 volts. The minimum battery discharge voltage (requiring opening the degraded battery output breaker) is 105 VDC (ref. 1, 6).

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

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 8.0 Electrical Power
2. DB-OP-02537 Loss of D1P and DAP
3. DB-OP-02538 Loss of D2P and DBP
4. DB-OP-02539 Loss of D1N and DAN
5. DB-OP-02540 Loss of D2N and DBN
6. System Description for 125/250 VDC and 120 V Instrumentation AC System
7. 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-4 onsite communication methods

Table C-4 Communication Methods

System	Onsite	ORO	NRC
Public Address (Gaitronics)	X		
Onsite Radios	X		
Plant Telephones	X	X	X
Commercial Telephones	X	X	X
4-Way Ringdown Circuit		X	
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System (ETS)			X

Mode Applicability:

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

NEI 99-01 Basis:

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

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

DBNPS Basis:

Onsite/Offsite Response Organization (ORO)/NRC communications include one or more of the systems listed in Table C-4 (ref. 1, 2).

Public Address (Gaitronics) System

The DBNPS public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

On-site Radio System

Radio systems can be used for communication among operators, off-site monitoring teams, the control room, security, TSC and EOF.

Plant Telephone System

The DBNPS plant telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code as well as external or offsite calling capability.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by DBNPS. The local service provider provides primary and secondary power for their lines at the Central Office.

4-Way Ringdown Circuit

Dedicated ring down line that includes the State and County EOCs, the Ohio Highway Patrol Office, the Lucas County and Ottawa County Sheriff's dispatcher offices, the Emergency Operations Facility, and the Control Room.

Satellite Phones

Portable satellite telephones are available which enable communication when all other phone systems are inoperable, e.g. following a major external event. Internal batteries, external DC sources as well as external AC sources can power these portable systems.

Cellular Phones

Cellular phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telephone System

The NRC uses a DBNPS dedicated telephone line, which allows direct telephone communications from the plant to NRC regional and national offices. The DBNPS communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the DBNPS Control Room, Technical Support Center, and Emergency Operations Facility and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

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

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 9.5.2 Communications Systems
2. DBNPS Emergency Plan 7.6 Communications Systems
3. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.2 Unusual Event

Loss of **ALL** Table C-4 ORO communication methods

Table C-4 Communication Methods

System	Onsite	ORO	NRC
Public Address (Gaitronics)	X		
Onsite Radios	X		
Plant Telephones	X	X	X
Commercial Telephones	X	X	X
4-Way Ringdown Circuit		X	
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System (ETS)			X

Mode Applicability:

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

NEI 99-01 Basis:

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

This EAL 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, Lucas and Ottawa County EOCs.

DBNPS Basis:

Onsite/Offsite Response Organization (ORO)/NRC communications include one or more of the systems listed in Table C-4 (ref. 1, 2).

Public Address (Gaitronics) System

The DBNPS public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

On-site Radio System

Radio systems can be used for communication among operators, off-site monitoring teams, the control room, security, TSC and EOF.

Plant Telephone System

The DBNPS plant telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code as well as external or offsite calling capability.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by DBNPS. The local service provider provides primary and secondary power for their lines at the Central Office.

4-Way Ringdown Circuit

Dedicated ring down line that includes the State and County EOCs, the Ohio Highway Patrol Office, the Lucas County and Ottawa County Sheriff's dispatcher offices, the Emergency Operations Facility, and the Control Room.

Satellite Phones

Portable satellite telephones are available which enable communication when all other phone systems are inoperable, e.g. following a major external event. Internal batteries, external DC sources as well as external AC sources can power these portable systems.

Cellular Phones

Cellular phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telephone System

The NRC uses a DBNPS dedicated telephone line, which allows direct telephone communications from the plant to NRC regional and national offices. The DBNPS communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the DBNPS Control Room, Technical Support Center, and Emergency Operations Facility and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

This EAL is the cold condition equivalent of the hot condition EAL SU7.2.

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 9.5.2 Communications Systems
2. DBNPS Emergency Plan 7.6 Communications Systems
3. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.3 Unusual Event

Loss of **ALL** Table C-4 NRC communication methods

Table C-4 Communication Methods

System	Onsite	ORO	NRC
Public Address (Gaitronics)	X		
Onsite Radios	X		
Plant Telephones	X	X	X
Commercial Telephones	X	X	X
4-Way Ringdown Circuit		X	
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System (ETS)			X

Mode Applicability:

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

NEI 99-01 Basis:

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

This EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

DBNPS Basis:

Onsite/Offsite Response Organization (ORO)/NRC communications include one or more of the systems listed in Table C-4 (ref. 1, 2).

Public Address (Gaitronics) System

The DBNPS public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

On-site Radio System

Radio systems can be used for communication among operators, off-site monitoring teams, the control room, security, TSC and EOF.

Plant Telephone System

The DBNPS plant telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code as well as external or offsite calling capability.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by DBNPS. The local service provider provides primary and secondary power for their lines at the Central Office.

4-Way Ringdown Circuit

Dedicated ring down line that includes the State and County EOCs, the Ohio Highway Patrol Office, the Lucas County and Ottawa County Sheriff's dispatcher offices, the Emergency Operations Facility, and the Control Room.

Satellite Phones

Portable satellite telephones are available which enable communication when all other phone systems are inoperable, e.g. following a major external event. Internal batteries, external DC sources as well as external AC sources can power these portable systems.

Cellular Phones

Cellular phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telephone System

The NRC uses a DBNPS dedicated telephone line, which allows direct telephone communications from the plant to NRC regional and national offices. The DBNPS communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the DBNPS Control Room, Technical Support Center, and Emergency Operations Facility and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

This EAL is the cold condition equivalent of the hot condition EAL SU7.3.

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 9.5.2 Communications Systems
2. DBNPS Emergency Plan 7.6 Communications Systems
3. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting SAFETY SYSTEMS

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

EAL:

CA6.1 Alert

The occurrence of **ANY** Table C-5 Hazardous Event

AND EITHER:

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

Table C-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Emergency Director

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

NEI 99-01 Basis:

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

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

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

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

DBNPS Basis:

- Ground motion acceleration of 0.08g horizontal or 0.053g vertical is the Maximum Probable Earthquake as is considered generically as the Operating Basis Earthquake for DBNPS (ref. 8). Control room alarm indication of an earthquake greater than OBE is indicated on the seismic control panel (C5764A). RA-EP-02820 Earthquake provides the guidance for determining any required response actions if the OBE earthquake threshold is exceeded (ref. 1). The significance of seismic events is discussed under EAL HU2.1.

- Internal FLOODING occurs from breaches of water systems that are located inside plant buildings and are connected to large water sources such as Intake Forebay or tanks (ref. 2).

- External FLOODING may be due to high lake level. DBNPS flood emergency elevation is 578 ft. Site access would be limited to rail, boat or helicopter (ref. 3).

- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 90 mph. (ref. 4).

- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area in the fire abnormal procedure (ref. 5).

- An EXPLOSION that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL (ref. 6).

DBNPS Basis Reference(s):

1. RA-EP-02820 Earthquake
2. RA-EP-02880 Internal Flooding
3. RA-EP-02830 Flooding
4. DBNPS UFSAR Section 3.3.1 Wind Criteria
5. DB-OP-02501 Serious Station Fire
6. RA-EP-02840 Explosion

7. NEI 99-01 CA6
8. Updated FSAR Section 3.1 Seismic Design

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 Essential AC Power

Loss of emergency electrical power can compromise plant SAFETY SYSTEM operability including decay heat removal and emergency core cooling systems, which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite sources for 4160 V essential buses.

2. Loss of Essential DC Power

Loss of emergency electrical power can compromise plant SAFETY SYSTEM operability including decay heat removal and emergency core cooling systems, which may be necessary to ensure fission product barrier integrity. This category includes loss of essential plant 125 VDC power sources.

3. Loss of Control Room Indications

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

4. RCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant increase from these base-line levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

5. RCS Leakage

The reactor vessel provides a volume for the coolant that covers the reactor core. The reactor pressure vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad

integrity fail. Excessive RCS leakage greater than Technical Specification limits indicates potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and containment integrity.

6. RPS Failure

This subcategory includes events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RPS to complete a reactor trip comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean ANY trip failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and containment integrity.

7. Loss of Communications

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

8. Containment Failure

Failure of containment isolation capability (under conditions in which the containment is not currently challenged) warrants emergency classification. Failure of containment pressure control capability also warrants emergency classification.

9. Hazardous Event Affecting SAFETY SYSTEMS

Various natural and technological events that result in degraded plant SAFETY SYSTEM performance or significant visible damage warrant emergency classification under this subcategory.

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Prolonged loss of **ALL** offsite and **ALL** onsite AC power to essential buses

EAL:

SG1.1 General Emergency

Loss of **ALL** offsite and **ALL** onsite AC power capability, Table S-1, to essential 4160V buses C1 and D1

AND EITHER:

- Restoration of at least one essential bus in < 4 hours is **not** likely (Note 1)
- Calculated Clad Temperature in **Region 3** (DB-OP-02000 Figure 2)

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

Table S-1 Offsite/Onsite AC Power Sources
Offsite: • X11 • X11 (back-fed via Main Transformer) • X01 • X02 **Onsite:** • EDG1 • EDG2 • SBODG

Mode Applicability:

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

NEI 99-01 Basis:

This IC addresses a prolonged loss of ALL power sources to AC essential buses. A loss of ALL AC power compromises the performance of ALL SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A prolonged

loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

The EAL should require declaration of a General Emergency prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC essential bus by the end of the analyzed station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of challenges to multiple fission product barriers.

The estimate for restoring at least one essential 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.

DBNPS Basis:

This EAL is indicated by the extended loss of ALL offsite and onsite AC power capability to 4160V essential buses C1 and D1 either for greater than 4 hours or that has resulted in indications of an actual loss of adequate core cooling.

Indication of continuing core cooling degradation is manifested by Calculated Clad Temperature in Region 3 (DB-OP-02000 Figure 2) (ref. 3).

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses C1 and D1 (ref. 1).

The essential buses during plant operation are normally powered from the 13.8KV offsite power system through their respective 13.8KV/4160V bus tie transformers, via the Unit Auxiliary Transformer (X11). In non-power operating modes, the essential buses may be back-fed via the X11 and Main Transformer provided the main generator lead disconnect links are removed (ref. 1, 2).

A standby source of offsite power to each 4160V essential bus is provided from the 13.8KV offsite power system via two separate and independent 13.8KV/4160V Startup Transformers (X01 & X02). Normally each startup transformer serves as a reserve power source for only one essential bus. However, a single startup transformer can be aligned to power both essential buses (ref. 1, 2).

Each essential bus has a diesel generator (EDG1 & EDG2) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power (ref. 1).

An Alternate AC power source, the Station Black Out (SBO) Diesel Generator, is located onsite. This onsite AC power source can be started from the Control Room and be loaded within 10 minutes of a SBO (ref. 1).

The SBODG fuel oil supply is separate from fuel oil supply for the station's EDGs. The SBODG fuel oil supply tank capacity is based on an eight-hour supply with the SBODG at rated load. A minimum supply of four hours must be stored in this tank to meet the site's station blackout duration analysis conditions (ref 1).

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on Emergency Director judgment as it relates to IMMINENT Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers. Indication of continuing core cooling degradation is manifested by Calculated Clad Temperature in Region 3 (DB-OP-02000 Figure 2). Figure 2, Incore T/C Temperature vs. RCS Pressure for ICC, provides indication of how serious core conditions are based upon combinations of RCS pressure and incore thermocouple temperatures. If the RCS P-T point is in Region 3, the cladding temperatures in the high power regions of the Core may be 1400ºF or higher. This is a serious condition, which could lead to significant amounts of H_2 production; Core damage may be unavoidable (ref. 3, 4).

DBNPS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. DB-OP-02521 Loss of AC Bus Power Sources
3. DB-OP-02000 Figure 2, Incore T/C Temperature vs. RC Pressure for Inadequate Core Cooling
4. Bases and Deviation Document for DB-OP-02000
5. NEI 99-01 SG1

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of **ALL** essential AC and DC power sources for 15 minutes or longer

EAL:

SG1.2 General Emergency

Loss of **ALL** offsite and **ALL** onsite AC power capability, Table S-1, to essential 4160V buses C1 and D1 for ≥ 15 min.

AND

Loss of **ALL** 125 VDC power based on battery bus voltage indications < 105 VDC on **ALL** essential DC distribution panels D1P, D1N, D2P and D2N for ≥ 15 min.

(Note 1)

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

Table S-1 Offsite/Onsite AC Power Sources
Offsite: • X11 • X11 (back-fed via Main Transformer) • X01 • X02 **Onsite:** • EDG1 • EDG2 • SBODG

Mode Applicability:

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

NEI 99-01 Basis:

This IC addresses a concurrent and prolonged loss of both essential AC AND DC power. A loss of ALL essential 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 essential DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both AC and 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.

DBNPS Basis:

This EAL is indicated by the loss of ALL offsite and onsite essential AC power capability to 4160V essential buses C1 and D1 for greater than 15 minutes in combination with degraded DC power voltage. This EAL addresses operating experience from the March 2011 accident at Fukushima Daiichi.

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses C1 and D1 (ref. 1).

The essential buses during plant operation are normally powered from the 13.8KV offsite power system through their respective 13.8KV/4160V bus tie transformers, via the Unit Auxiliary Transformer (X11). In non-power operating modes, the essential buses may be back-fed via the X11 and Main Transformer provided the main generator lead disconnect links are removed (ref. 1, 2). Credit for the X11 back-feed can only be taken if already aligned, as it takes greater than 15 minutes to align.

A standby source of offsite power to each 4160V essential bus is provided from the 13.8KV offsite power system via two separate and independent 13.8KV/4160V Startup Transformers (X01 & X02). Normally each startup transformer serves as a reserve power source for only one essential bus. However, a single startup transformer can be aligned to power both essential buses (ref. 1, 2).

Each essential bus has a diesel generator (EDG1 & EDG2) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power (ref. 1).

An Alternate AC power source, the Station Black Out (SBO) Diesel Generator, is located onsite. This onsite AC power source can be started from the Control Room and be loaded within 10 minutes of a SBO (ref. 1).

The DC electrical distribution subsystem consists of two trains, designated Train 1 and Train 2. Each train consists of a 250/125 VDC motor control center (MCC), and each 250/125 VDC MCC consists of two 125 VDC buses (one positive and one negative) that can be powered

from a polarity specific battery or battery charger. The two DC buses then supply a specific Essential DC Distribution Panel (D1P and D1N for DCMCC 1, and D2P and D2N for DCMCC 2). The four Essential DC Distribution Panels independently supply DC electrical power to the four inverters, which in turn power the 120 VAC essential buses (ref. 1, 3, 4, 5, 6, 7).

The Class 1E DC loads have an operating voltage range of 105 to 135 volts. The minimum battery discharge voltage (requiring opening the degraded battery output breaker) is 105 VDC (ref. 1, 7).

DBNPS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. DB-OP-02521 Loss of AC Bus Power Sources
3. DB-OP-02537 Loss of D1P and DAP
4. DB-OP-02538 Loss of D2P and DBP
5. DB-OP-02539 Loss of D1N and DAN
6. DB-OP-02540 Loss of D2N and DBN
7. System Description for 125/250 VDC and 120 V Instrumentation AC System
8. NEI 99-01 SG8

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of **ALL** offsite power and **ALL** onsite AC power to essential 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 essential 4160V buses C1 and D1 for ≥ 15 min. (Note 1)

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

Table S-1 Offsite/Onsite AC Power Sources
Offsite: • X11 • X11 (back-fed via Main Transformer) • X01 • X02 **Onsite:** • EDG1 • EDG2 • SBODG

Mode Applicability:

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

NEI 99-01 Basis:

This IC addresses a total loss of AC power that compromises the performance of ALL SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

DBNPS Basis:

This EAL is indicated by the loss of ALL offsite and onsite AC power capability (Table S-1) to 4160V essential buses C1 and D1. The essential switchgear are buses C1 (Train A) and D1 (Train B) (ref. 1). For emergency classification purposes, “capability” means that an AC power source is available to the essential buses, whether or not the buses are powered from it.

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses C1 and D1 (ref. 1).

The essential buses during plant operation are normally powered from the 13.8KV offsite power system through their respective 13.8KV/4160V bus tie transformers, via the Unit Auxiliary Transformer (X11). In non-power operating modes, the essential buses may be back-fed via the X11 and Main Transformer provided the main generator lead disconnect links are removed (ref. 1, 2). Credit for the X11 back-feed can only be taken if already aligned, as it takes greater than 15 minutes to align.

A standby source of offsite power to each 4160V essential bus is provided from the 13.8KV offsite power system via two separate and independent 13.8KV/4160V Startup Transformers (X01 & X02). Normally each startup transformer serves as a reserve power source for only one essential bus. However, a single startup transformer can be aligned to power both essential buses (ref. 1, 2).

Each essential bus has a diesel generator (EDG1 & EDG2) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power (ref. 1).

An Alternate AC power source, the Station Black Out (SBO) Diesel Generator, is located onsite. This onsite AC power source can be started from the Control Room and be loaded within 10 minutes of a SBO (ref. 1).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. The interval begins when both offsite and onsite AC power capability are lost.

DBNPS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. DB-OP-02521 Loss of AC Bus Power Sources
3. NEI 99-01 SS1

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

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

EAL:

SA1.1 Alert

AC power capability, Table S-1, to essential 4160V buses C1 and D1 reduced to a single power source for ≥ 15 min. (Note 1)

AND

ANY additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS

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

Table S-1 Offsite/Onsite AC Power Sources
Offsite: • X11 • X11 (back-fed via Main Transformer) • X01 • X02 **Onsite:** • EDG1 • EDG2 • SBODG

Mode Applicability:

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

Basis:

NEI 99-01 Basis:

This IC describes a significant degradation of offsite and onsite AC power sources such that ANY additional single failure would result in a loss of ALL AC power to SAFETY SYSTEMS.

In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

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

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

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

Escalation of the emergency classification level would be via IC SS1.

DBNPS Basis:

For emergency classification purposes, “capability” means that an AC power source is available to the essential buses, whether or not the buses are powered from it.

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses C1 and D1 (ref. 1).

The essential buses during plant operation are normally powered from the 13.8KV offsite power system through their respective 13.8KV/4160V bus tie transformers, via the Unit Auxiliary Transformer (X11). In non-power operating modes, the essential buses may be back-fed via the X11 and Main Transformer provided the main generator lead disconnect links are removed (ref. 1, 2). Credit for the X11 back-feed can only be taken if already aligned, as it takes greater than 15 minutes to align.

A standby source of offsite power to each 4160V essential bus is provided from the 13.8KV offsite power system via two separate and independent 13.8KV/4160V Startup Transformers (X01 & X02). Normally each startup transformer serves as a reserve power source for only one essential bus. However, a single startup transformer can be aligned to power both essential buses (ref. 1, 2).

Each essential bus has a diesel generator (EDG1 & EDG2) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power (ref. 1).

An Alternate AC power source, the Station Black Out (SBO) Diesel Generator, is located onsite. This onsite AC power source can be started from the Control Room and be loaded within 10 minutes of a SBO (ref. 1).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. If the capability of a second source of essential bus power is not restored within 15 minutes, an Alert is declared under this EAL.

DBNPS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. DB-OP-02521 Loss of AC Bus Power Sources
3. NEI 99-01 SA1

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

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

EAL:

SU1.1 Unusual Event

Loss of **ALL** offsite AC power capability, Table S-1, to essential 4160V buses C1 and D1 for ≥ 15 min. (Note 1)

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

Table S-1 Offsite/Onsite AC Power Sources
Offsite: • X11 • X11 (back-fed via Main Transformer) • X01 • X02 **Onsite:** • EDG1 • EDG2 • SBODG

Mode Applicability:

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

Basis:

NEI 99-01 Basis:

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

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

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

Escalation of the emergency classification level would be via IC SA1.

DBNPS Basis:

For emergency classification purposes, “capability” means that an AC power source is available to the essential buses, whether or not the buses are powered from it.

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses C1 and D1 (ref. 1).

The essential buses during plant operation are normally powered from the 13.8KV offsite power system through their respective 13.8KV/4160V bus tie transformers, via the Unit Auxiliary Transformer (X11). In non-power operating modes, the essential buses may be back-fed via the X11 and Main Transformer provided the main generator lead disconnect links are removed (ref. 1, 2). Credit for the X11 back-feed can only be taken if already aligned, as it takes greater than 15 minutes to align.

A standby source of offsite power to each 4160V essential bus is provided from the 13.8KV offsite power system via two separate and independent 13.8KV/4160V Startup Transformers (X01 & X02). Normally each startup transformer serves as a reserve power source for only one essential bus. However, a single startup transformer can be aligned to power both essential buses (ref. 1, 2).

Each essential bus has a diesel generator (EDG1 & EDG2) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power (ref. 1).

An Alternate AC power source, the Station Black Out (SBO) Diesel Generator, is located onsite. This onsite AC power source can be started from the Control Room and be loaded within 10 minutes of a SBO (ref. 1).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses.

DBNPS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. DB-OP-02521 Loss of AC Bus Power Sources
3. NEI 99-01 SU1

Category: S – System Malfunction

Subcategory: 2 – Loss of Essential DC Power

Initiating Condition: Loss of **ALL** essential 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** essential DC distribution panels D1P, D1N, D2P and D2N for ≥ 15 min. (Note 1)

Note 1: The Emergency Director 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

NEI 99-01 Basis:

This IC addresses a loss of essential 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.

DBNPS Basis:

The DC electrical distribution subsystem consists of two trains, designated Train 1 and Train 2. Each train consists of a 250/125 VDC motor control center (MCC), and each 250/125 VDC MCC consists of two 125 VDC buses (one positive and one negative) that can be powered from a polarity specific battery or battery charger. The two DC buses then supply a specific Essential DC Distribution Panel (D1P and D1N for DCMCC 1, and D2P and D2N for DCMCC 2). The four Essential DC Distribution Panels independently supply DC electrical power to the four inverters, which in turn power the 120 VAC essential buses (ref. 1, 2, 3, 4, 5, 6).

The Class 1E DC loads have an operating voltage range of 105 to 135 volts. The minimum battery discharge voltage (requiring opening the degraded battery output breaker) is 105 VDC (ref. 1, 6).

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 8.0 Electrical Power
2. DB-OP-02537 Loss of D1P and DAP
3. DB-OP-02538 Loss of D2P and DBP
4. DB-OP-02539 Loss of D1N and DAN
5. DB-OP-02540 Loss of D2N and DBN
6. System Description for 125/250 VDC and 120 V Instrumentation AC System
7. NEI 99-01 SS8

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer 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 SAFETY SYSTEM parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

ANY Significant Transient is or may be in progress, Table S-3

Note 1: The Emergency Director 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 pressure • In-core T/C temperature • Level in at least one S/G • Auxiliary or emergency feed flow

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

Mode Applicability:

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

NEI 99-01 Basis:

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an “inability to monitor” means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of ALL of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if ALL indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RCS pressure cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via ICs FS1 or RS1

DBNPS Basis:

SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Process Computer, which displays SPDS required information, serves as a redundant compensatory indicator, which may be utilized in lieu of normal Control Room indicators (ref. 1, 2, 3).

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 25% thermal power

change, electrical load rejections of greater than 25% full electrical load or SI injection actuations.

DBNPS Basis Reference(s):

1. UFSAR Section 7.5 Safety-Related Display Instrumentation
2. DB-OP-02541 Loss of YAU
3. DB-OP-02542 Loss of YBU
4. NEI 99-01 SA2

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 SAFETY SYSTEM parameters from within the Control Room for ≥ 15 min. (Note 1)

Note 1: The Emergency Director 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 pressure • In-core T/C temperature • Level in at least one S/G • Auxiliary or emergency feed flow

Mode Applicability:

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

NEI 99-01 Basis:

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an “inability to monitor” means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of ALL of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating

procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if ALL indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RCS pressure cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via IC SA3.

DBNPS Basis:

SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Process Computer, which displays SPDS required information, serves as a redundant compensatory indicator, which may be utilized in lieu of normal Control Room indicators (ref. 1, 2, 3).

DBNPS Basis Reference(s):

1. UFSAR Section 7.5 Safety-Related Display Instrumentation
2. DB-OP-02541 Loss of YAU
3. DB-OP-02542 Loss of YBU
4. NEI 99-01 SU2

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.1	**Unusual Event**
Letdown Monitor (RE 1998) reading > 2.0E+06 cpm	

Mode Applicability:

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

NEI 99-01 Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

DBNPS Basis:

The DBNPS historical highest failed fuel level of 0.007% corresponds to a reading of 1.4E+05 cpm on the Failed Fuel Monitor (RE 1998). The top of scale for RE 1998 is 1.0E+07. A monitor value of 2.0E+06 (0.1% clad damage) was chosen for its ability to be recognized even though exceeding Technical Specification Limits could potentially result in much higher readings (ref. 1, 2).

DBNPS Basis Reference(s):

1. Radiation Monitor Setpoint Manual, RE 1998 Failed Fuel Detector
2. DBNPS Technical Specifications LCO 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

RCS activity > Technical Specification LCO 3.4.16 as indicated by **ANY** of the following:

- Dose equivalent I-131 in the unacceptable region of Figure 3.4.16-1
- > 1 µCi/gm dose equivalent I-131 for > 48 hrs
- > 100/Ē µCi/gm gross specific coolant activity

Mode Applicability:

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

NEI 99-01 Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

DBNPS Basis:

Technical Specification LCO 3.4.16 Condition A limits RCS System Dose Equivalent I-131 to ≤ 1.0 µCi/gm for > 48 hours. Technical Specification Section 3.4.16 Condition B limits RCS System Dose Equivalent I-131 to within Figure 3.4.16-1 and gross specific activity to ≤ 100/Ē µCi/gm. (ref 1).

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications section 3.4.16 RCS Specific Activity
2. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1	**Unusual Event**
RCS leakage > 10 gpm for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director 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

NEI 99-01 Basis:

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.

This EAL is focused on a loss of mass from the RCS due to “unidentified leakage" or "pressure boundary leakage" (as these leakage types are defined in the plant Technical Specifications).

The leak rate value was selected because it is usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation).

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.

DBNPS Basis:

Unidentified leakage is ALL leakage (except RCP seal return) that is not identified leakage (ref. 1, 2).

DBNPS does not have the capability to classify leakage as identified leakage within 15 minutes. Therefore, for the purpose of this IC and EAL all RCS leakage is considered unidentified leakage and the 10 gpm leak rate applies.

Escalation of this EAL to the Alert level is via Category F, Fission Product Barrier Degradation, EAL FA1.1.

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications Definitions section 1.1
2. UFSAR Section 5.2.4.7 Leakage Identification
3. DB-OP-02522 Small RCS Leaks
4. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%

AND

ALL actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5%

AND EITHER:

- Calculated Clad Temperature in **Region 3** (DB-OP-02000 Figure 2)
- MFW, AFW and MU-HPI PORV Cooling are **all** unavailable

Mode Applicability:

1 - Power Operation

NEI 99-01 Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, ALL subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shut down the reactor. The inclusion of this IC and EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Escalation of the emergency classification level would be via ICs RG1 or FG1.

DBNPS Basis:

This EAL addresses the following:

- ANY automatic reactor trip signal followed by a manual trip that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the SAFETY SYSTEMS were designed (EAL SA6.1), AND

- Indications that either core cooling is extremely challenged or heat removal is extremely challenged

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS barriers.

Reactor shutdown achieved by use of other trip actions specified in DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture (locally opening Reactor Trip Breaker, emergency boration or manually driving control rods) are also credited as a successful manual trip methods provided reactor power can be reduced below 5% before indications of an extreme challenge to either core cooling or heat removal exist (ref. 1).

Each of the redundant Auxiliary Feed Pumps is sized to provide 100% of the capacity required by the SGs to remove 5% of the reactor thermal power produced at full load steam pressure conditions. The ≥ 5% power portion of this EAL threshold was chosen to be an easily recognizable, onscale indication that the reactor trip has not functioned to shutdown the reactor assuming the reactor power was in the normal operating range.

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on Emergency Director judgment as it relates to IMMINENT Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers. Indication of continuing core cooling degradation is manifested by Calculated Clad Temperature in Region 3 (DB-OP-02000 Figure 2). Figure 2, Incore T/C Temperature vs. RCS Pressure for ICC, provides indication of how serious core conditions are based upon combinations of RCS pressure and incore thermocouple temperatures. If the RCS P-T point is in Region 3, the cladding temperatures in the high power regions of the core may be 1400°F or higher. This is a serious condition, which will lead to significant amounts of H2 production; core damage may be unavoidable (ref. 1).

An extreme challenge to heat removal is defined as a complete loss of MFW, AFW and MU-HPI PORV Cooling (ref. 2, 3).

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications section 3.3.1 Reactor Protection System (RPS) Instrumentation
2. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
3. Bases and Deviation Document for DB-OP-02000

4. NEI 99-01 SS5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor AND subsequent manual actions taken at the Controls Area 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 Controls Area (manual RPS trip pushbuttons and de-energizing E2 and F2) 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

NEI 99-01 Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, AND subsequent operator manual actions taken at the Controls Area to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the Controls Area since this event entails a significant failure of the RPS.

A manual action at the Controls Area 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 Controls Area (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 Controls Area”.

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant

conditions, etc. If the failure to shut down the reactor is prolonged enough to cause a challenge to the core cooling or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6 or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

DBNPS Basis:

This EAL addresses ANY automatic or manual reactor trip signal that fails to shut down the reactor 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.

For the purposes of emergency classification, successful manual trip actions are those, which can be quickly performed from the Controls Area (ATCA) (i.e., manual pushbuttons or de-energizing E2 and F2). Reactor shutdown achieved by use of other trip actions specified in DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture (locally opening Reactor Trip Breaker, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 1).

Each of the redundant Auxiliary Feed Pumps is sized to provide 100% of the capacity required by the SGs to remove 5% of the reactor thermal power produced at full load steam pressure conditions. The ≥ 5% power portion of this EAL threshold was chosen to be an easily recognizable, onscale indication that the reactor trip has not functioned to shutdown the reactor assuming the reactor power was in the normal operating range.

Escalation of this event to a Site Area Emergency would be under EAL SS6.1 or Emergency Director judgment.

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

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

EAL:

SU6.1 Unusual Event

An automatic trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **ANY** RPS setpoint is exceeded

AND

A subsequent manual trip action taken at the Controls Area (manual RPS trip pushbuttons or de-energizing E2 and F2) 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

NEI 99-01 Basis:

This EAL addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, AND a subsequent operator manual action taken at the Controls Area 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 Controls Area to shutdown the reactor (e.g., initiate a manual reactor trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the Controls Area to shutdown the reactor (e.g., initiate a manual reactor trip) using a different switch). Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the Controls Area is ANY operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This

action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be “at the Controls Area”.

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 Controls Area are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor trip signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the trip failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

DBNPS Basis:

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

Following a successful reactor trip, rapid insertion of the control rods occurs. Reactor power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from an automatic reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful trip has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2).

Each of the redundant Auxiliary Feed Pumps is sized to provide 100% of the capacity required by the SGs to remove 5% of the reactor thermal power produced at full load steam pressure conditions. The ≥ 5% power portion of this EAL threshold was chosen to be an easily

recognizable, onscale indication that the reactor trip has not functioned to shutdown the reactor assuming the reactor power was in the normal operating range.

For the purposes of emergency classification, successful manual trip actions are those, which can be quickly performed from the Controls Area (ATCA) (i.e., manual pushbuttons or de-energizing E2 and F2). Reactor shutdown achieved by use of other trip actions specified in DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture (locally opening Reactor Trip Breaker, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 2).

In the event that the operator identifies a reactor trip is imminent and initiates a successful manual reactor trip before the automatic RPS trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. However, if subsequent manual reactor trip actions fail to reduce reactor power below 5%, the event escalates to the Alert under EAL SA6.1.

If by procedure, operator actions include the initiation of an immediate manual trip following receipt of an automatic trip signal and there are no clear indications that the automatic trip failed (such as a time delay following indications that a trip setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic trip or manual actions. If a subsequent review of the trip actuation indications reveals that, the automatic trip did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 10 CFR 50.72 should be considered for the transient event.

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications section 3.3.1 Reactor Protection System (RPS) Instrumentation
2. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
3. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

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

EAL:

SU6.2 Unusual Event

A manual trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **ANY** manual trip action was initiated

AND

A subsequent automatic trip or manual trip action taken at the Controls Area (manual RPS trip pushbuttons or de-energizing E2 and F2) 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

NEI 99-01 Basis:

This EAL addresses a failure of the RPS to initiate or complete a manual reactor trip that results in a reactor shutdown, AND either a subsequent operator manual action taken at the Controls Area 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 Controls Area to shutdown the reactor (e.g., initiate a manual reactor trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the Controls Area to shutdown the reactor (e.g., initiate a manual reactor trip) using a different switch. Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the Controls Area is ANY operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This

action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be “at the Controls Area”.

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 Controls Area are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor trip signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the trip failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

DBNPS Basis:

This EAL addresses a failure of a manually initiated trip in the absence of having exceeded an automatic RPS trip setpoint AND a subsequent automatic or manual trip is successful in shutting down the reactor (reactor power < 5%) (ref. 1).

Following a successful reactor trip, rapid insertion of the control rods occurs. Reactor 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 RPS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2).

Each of the redundant Auxiliary Feed Pumps is sized to provide 100% of the capacity required by the SGs to remove 5% of the reactor thermal power produced at full load steam pressure conditions. The ≥ 5% power portion of this EAL threshold was chosen to be an easily recognizable, onscale indication that the reactor trip has not functioned to shutdown the reactor assuming the reactor power was in the normal operating range.

For the purposes of emergency classification, successful manual trip actions are those, which can be quickly performed from the Controls Area (ATCA) (i.e., manual pushbuttons or de-energizing E2 and F2). Reactor shutdown achieved by use of other trip actions specified in DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture (locally opening Reactor Trip Breaker, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 2).

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

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications section 3.3.1 Reactor Protection System (RPS) Instrumentation
2. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
3. NEI 99-01 SU5

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

Table S-4 Communication Methods

System	Onsite	ORO	NRC
Public Address (Gaitronics)	X		
Onsite Radios	X		
Plant Telephones	X	X	X
Commercial Telephones	X	X	X
4-Way Ringdown Circuit		X	
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System			X

Mode Applicability:

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

NEI 99-01 Basis:

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

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

DBNPS Basis:

Onsite/Offsite Response Organization (ORO)/NRC communications include one or more of the systems listed in Table S-4 (ref. 1, 2).

Public Address (Gaitronics) System

The DBNPS public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

On-site Radio System

Radio systems can be used for communication among operators, off-site monitoring teams, the control room, security, TSC and EOF.

Plant Telephone System

The DBNPS plant telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code as well as external or offsite calling capability.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by DBNPS. The local service provider provides primary and secondary power for their lines at the Central Office.

4-Way Ringdown Circuit

Dedicated ring down line that includes the State and County EOCs, the Ohio Highway Patrol Office, the Lucas County and Ottawa County Sheriff's dispatcher offices, the Emergency Operations Facility, and the Control Room.

Satellite Phones

Portable satellite telephones are available which enable communication when all other phone systems are inoperable, e.g. following a major external event. Internal batteries, external DC sources as well as external AC sources can power these portable systems.

Cellular Phones

Cellular phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telephone System

The NRC uses a DBNPS dedicated telephone line, which allows direct telephone communications from the plant to NRC regional and national offices. The DBNPS communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the DBNPS Control Room, Technical Support Center, and Emergency Operations Facility and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 9.5.2 Communications Systems
2. DBNPS Emergency Plan 7.6 Communications Systems
3. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

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

EAL:

SU7.2 Unusual Event

Loss of **ALL** Table S-4 ORO communication methods

Table S-4 Communication Methods

System	Onsite	ORO	NRC
Public Address (Gaitronics)	X		
Onsite Radios	X		
Plant Telephones	X	X	X
Commercial Telephones	X	X	X
4-Way Ringdown Circuit		X	
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System			X

Mode Applicability:

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

NEI 99-01 Basis:

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

This EAL 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 of Ohio, Lucas and Ottawa County EOCs.

DBNPS Basis:

Onsite/Offsite Response Organization (ORO)/NRC communications include one or more of the systems listed in Table S-4 (ref. 1, 2).

Public Address (Gaitronics) System

The DBNPS public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

On-site Radio System

Radio systems can be used for communication among operators, off-site monitoring teams, the control room, security, TSC and EOF.

Plant Telephone System

The DBNPS plant telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code as well as external or offsite calling capability.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by DBNPS. The local service provider provides primary and secondary power for their lines at the Central Office.

4-Way Ringdown Circuit

Dedicated ring down line that includes the State and County EOCs, the Ohio Highway Patrol Office, the Lucas County and Ottawa County Sheriff's dispatcher offices, the Emergency Operations Facility, and the Control Room.

Satellite Phones

Portable satellite telephones are available which enable communication when ALL other phone systems are inoperable, e.g. following a major external event. Internal batteries, external DC sources as well as external AC sources can power these portable systems.

Cellular Phones

Cellular phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telephone System

The NRC uses a DBNPS dedicated telephone line, which allows direct telephone communications from the plant to NRC regional and national offices. The DBNPS communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the DBNPS Control Room, Technical Support Center, and Emergency Operations Facility and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

This EAL is the hot condition equivalent of the cold condition EAL CU5.2.

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 9.5.2 Communications Systems
2. DBNPS Emergency Plan 7.6 Communications Systems
3. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

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

EAL:

SU7.3 Unusual Event

Loss of **ALL** Table S-4 NRC communication methods

Table S-4 Communication Methods

System	Onsite	ORO	NRC
Public Address (Gaitronics)	X		
Onsite Radios	X		
Plant Telephones	X	X	X
Commercial Telephones	X	X	X
4-Way Ringdown Circuit		X	
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System			X

Mode Applicability:

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

NEI 99-01 Basis:

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

This EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

DBNPS Basis:

Onsite/Offsite Response Organization (ORO)/NRC communications include one or more of the systems listed in Table S-4 (ref. 1, 2).

Public Address (Gaitronics) System

The DBNPS public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

On-site Radio System

Radio systems can be used for communication among operators, off-site monitoring teams, the control room, security, TSC and EOF.

Plant Telephone System

The DBNPS plant telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code as well as external or offsite calling capability.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by DBNPS. The local service provider provides primary and secondary power for their lines at the Central Office.

4-Way Ringdown Circuit

Dedicated ring down line that includes the State and County EOCs, the Ohio Highway Patrol Office, the Lucas County and Ottawa County Sheriff's dispatcher offices, the Emergency Operations Facility, and the Control Room.

Satellite Phones

Portable satellite telephones are available which enable communication when ALL other phone systems are inoperable, e.g. following a major external event. Internal batteries, external DC sources as well as external AC sources can power these portable systems.

Cellular Phones

Cellular phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telephone System

The NRC uses a DBNPS dedicated telephone line, which allows direct telephone communications from the plant to NRC regional and national offices. The DBNPS communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the DBNPS Control Room, Technical Support Center, and Emergency Operations Facility and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

This EAL is the hot condition equivalent of the cold condition EAL CU5.3.

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 9.5.2 Communications Systems
2. DBNPS Emergency Plan 7.6 Communications Systems
3. 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 closed within 15 min. of a VALID containment isolation signal (Note 1)	

Note 1: The Emergency Director 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

NEI 99-01 Basis:

This EAL addresses a failure of one or more containment penetrations to automatically isolate (close) when required by an actuation signal. Absent challenges to another fission product barrier, this condition represents potential degradation of the level of safety of the plant.

For this EAL 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. 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.

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.

DBNPS Basis:

Successful closure of any one valve in a penetration line is sufficient to consider the penetration closed.

DBNPS Basis Reference(s):

1. NEI 99-01 SU7

Category: S – System Malfunction

Subcategory: 8 – Containment Failure

Initiating Condition: Failure to isolate containment or loss of containment pressure control

EAL:

SU8.2 Unusual Event

Containment pressure > 40 psia with < one full train of containment cooling, Table S-6, operating per design for ≥ 15 min. (Note 1)

Table S-6 Containment Cooling Full Train	
CT Spray Pumps	**CT Cooling Fans**
2	0
1	1
0	2

Note 1: The Emergency Director 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

NEI 99-01 Basis:

This EAL 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, this condition represents a potential degradation of the level of safety of the plant.

This EAL addresses a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment sprays or ice condenser 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.

DBNPS Basis:

The combination of Containment spray pumps and Containment cooling fan units considered to be a full train of containment cooling operating per design is shown in Table F-3 (ref. 1).

SFAS 2 actuation automatically initiates Containment Air Coolers upon exceeding the Containment pressure high setpoint of 18.7 psia or low RCS pressure of 1600 psig. SFAS Level 4 actuation automatically initiates Containment Spray upon exceeding the Containment pressure high-high setpoint of 40 (nominal) psia (ref. 2, 3).

DBNPS Basis Reference(s):

1. UFSAR Section 6.2.2. Containment Vessel Heat Removal Systems
2. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
3. DBNPS Technical Specifications Table 3.3.5-1 Safety Features Actuation System Instrumentation
4. 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 has caused indications of degraded performance in at least one train of a SAFETY SYSTEM required for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure required for the current operating mode

Table S-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Emergency Director

Mode Applicability:

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

NEI 99-01 Basis:

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

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

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

Escalation of the emergency classification level would be via IC FS1 or RS1.

DBNPS Basis:

- Ground motion acceleration of 0.08g horizontal or 0.053g vertical is the Maximum Probable Earthquake as is considered generically as the Operating Basis Earthquake for DBNPS (ref. 8). Control room alarm indication of an earthquake greater than OBE is indicated on the seismic control panel (C5764A). RA-EP-02820 Earthquake provides the guidance for determining any required response actions if the OBE earthquake threshold is exceeded (ref. 1). The significance of seismic events is discussed under EAL HU2.1.

- Internal FLOODING occurs from breaches of water systems that are located inside plant buildings and are connected to large water sources such as Intake Forebay or tanks (ref. 2).

- External FLOODING may be due to high lake level. DBNPS flood emergency elevation is 578 ft. Site access would be limited to rail, boat or helicopter (ref. 3).

- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 90 mph. (ref. 4).

- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area in the fire abnormal procedure (ref. 5).

- An EXPLOSION that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL (ref. 6).

DBNPS Basis Reference(s):

1. RA-EP-02820 Earthquake
2. RA-EP-02880 Internal Flooding
3. RA-EP-02830 Flooding
4. DBNPS UFSAR Section 3.3.1 Wind Criteria
5. DB-OP-02501 Serious Station Fire
6. RA-EP-02840 Explosion
7. NEI 99-01 SA9
8. Updated FSAR Section 3.1 Seismic Design

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

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 2). “Loss” and “Potential Loss” signify the relative damage and threat of damage to the barrier. “Loss” means the barrier no longer assures containment of radioactive materials. “Potential Loss” means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Alert:

ANY loss or ANY potential loss of EITHER Fuel Clad or RCS

Site Area Emergency:

Loss or potential loss of ANY two barriers

General Emergency:

Loss of ANY two barriers AND loss or potential loss of third barrier

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

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier.

- Unusual Event ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.

- For accident conditions involving a radiological release, evaluation of the FISSION PRODUCT BARRIER THRESHOLDS will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the FISSION PRODUCT BARRIER THRESHOLDS may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC RG1 has been exceeded.

- The FISSION PRODUCT BARRIER THRESHOLDS specified within a scheme reflect plant-specific DBNPS design and operating characteristics.

- As used in this category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside the containment, an interfacing system, or outside of the containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered RCS leakage.

- At the Site Area Emergency level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: 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

NEI 99-01 Basis:

None

DBNPS 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

DBNPS Basis Reference(s):

1. NEI 99-01 FG1

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

NEI 99-01 Basis:

None

DBNPS 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 assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, the Emergency Director would have greater assurance that escalation to a General Emergency is less IMMINENT.

DBNPS Basis Reference(s):

1. NEI 99-01 FS1

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

NEI 99-01 Basis:

None

DBNPS 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

DBNPS Basis Reference(s):

1. NEI 99-01 FA1

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. CT Radiation / RCS Activity

D. CT Integrity or Bypass

E. Emergency Director Judgment

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

Thresholds are assigned sequential numbers within each 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 “CT 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 (CT) 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 (SFAS) actuation required by **EITHER**: • UNISOLABLE RCS leakage • SG tube RUPTURE	1. Operation of a standby Makeup Pump (>250 gpm) is required by **EITHER:** • UNISOLABLE RCS leakage • SG tube leakage **OR** 2. PTS requirements invoked (SR5)	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B **Inadequate Heat Removal**	1. Calculated Clad Temperature in **Region 3** or higher (DB-OP-02000 Figure 2)	1. Calculated Clad Temperature in **Region 2** or higher (DB-OP-02000 Figure 2) **OR** 2. Loss of **ALL** feedwater **AND** SG Cooling is required	None	1. Loss of **ALL** feedwater **AND** SG Cooling is required	None	1. Calculated Clad Temperature in **Region 3** or higher (DB-OP-02000 Figure 2) **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C **CT Radiation / RCS Activity**	1. RE 4596A or B > Table F-2 column "FC Loss" (Note 9) **OR** 2. Dose equivalent I-131 coolant activity > 300 µCi/gm	None	1. RE 4596A or B > Table F-2 column "RCS Loss" (Note 9)	None	None	1. RE 4596A or B > Table F-2 column "CT Potential Loss" (Note 9)
D **CT Integrity or Bypass**	None	None	None	None	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on Emergency Director judgment • UNISOLABLE pathway from Containment to the environment exists **OR** 2. Indications of RCS leakage outside of containment	1. Containment pressure > 50.4 psia **OR** 2. Containment hydrogen concentration > 4% **OR** 3. Containment pressure > 40 psia with < one full train, Table F-3, of containment cooling operating per design for ≥ 15 min. (Note 1)
E **ED Judgment**	1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier

Table F-2 Containment Radiation – R/hr (RE 4596A or B)			
Time After S/D (Hrs.)	**RCS Loss**	**FC Loss**	**CT Potential Loss**
0-1	1.50E+01	3.03E+03	1.40E+04
1-2	1.50E+01	2.56 E+03	1.18 E+04
2-8	1.50E+01	1.61 E+03	7.46 E+03
8-16	1.50E+01	1.14 E+03	5.28 E+03
16-24	1.50E+01	8.66 E+02	4.00 E+03
>24	1.50E+01	3.94 E+02	1.82 E+03

Table F-3 Containment Cooling Full Train	
Spray	**Coolers**
2	0
1	1
0	2

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. Calculated Clad Temperature in **Region 3** or higher (DB-OP-02000 Figure 2)

Basis:

Generic

This reading indicates temperatures within the core have caused significant superheating of reactor coolant.

Plant-Specific

Indication of severe core cooling degradation is manifested by Calculated Clad Temperature in Region 3 or higher (DB-OP-02000 Figure 2). Figure 2, Incore T/C Temperature vs. RCS Pressure for ICC, provides indication of how serious core conditions are based upon combinations of RCS pressure and incore thermocouple temperatures. If the RCS P-T point is in Region 3, the cladding temperatures in the core may be 1400°F or higher. This is a very serious condition and may lead to significant amounts of H2 production; core damage may be unavoidable as this represents a very serious inadequate core cooling condition (ref. 1, 2).

WCAP-14969-A states, "Analyses performed for the WOG ERGs for indication of inadequate core cooling concluded that the temperature indicated by the core exit thermocouples, especially during transient heatup conditions, is always several hundred degrees lower than the fuel rod cladding temperatures. Thus, an indicated temperature of 1200°F can be translated to a peak cladding temperature on the order of 1400°F" (ref. 4).

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. Bases and Deviation Document for DB-OP-02000
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A
4. WCAP-14969-A, Westinghouse Owners Group Core Damage Assessment Guidance

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. Calculated Clad Temperature in **Region 2** or higher (DB-OP-02000 Figure 2)

Basis:

Generic

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

Plant-Specific

The average incore thermocouple temperature and RCS pressure is used to determine whether Calculated Clad Temperature is in Region 2. This corresponds to a loss of RCS subcooling with clad temperatures remaining below the point where damage is immediately likely (T_{clad} approximately 900° to 1100° F).

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. Bases and Deviation Document for DB-OP-02000
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Potential Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. Loss of **ALL** feedwater **AND** SG cooling is required

Basis:

Generic

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.

Plant-Specific

In combination with RCS Potential Loss B.1, meeting this threshold would result in a Site Area Emergency.

Loss of ALL feedwater cooling heat transfer capability when SG cooling is required indicates the ultimate heat sink function is under extreme challenge AND that the RCS barrier is also challenged (ref. 1).

The phrase AND “SG cooling is required” precludes the need for classification for conditions in which RCS pressure is less than SG pressure. For large LOCA events inside the Containment, the SGs are moot because heat removal through the containment heat removal systems takes place. Therefore, SG cooling should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification (ref. 1, 2).

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. Bases and Deviation Document for DB-OP-02000
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.B

Barrier: Fuel Clad

Category: C. CT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. RE 4596A or B > Table F-2 column "FC Loss" (Note 9)

Table F-2 Containment Radiation – R/hr (RE 4596A or B)			
Time After S/D (Hrs.)	**RCS Loss**	**FC Loss**	**CT Potential Loss**
0-1	1.50E+01	3.03E+03	1.40E+04
1-2	1.50E+01	2.56 E+03	1.18 E+04
2-8	1.50E+01	1.61 E+03	7.46 E+03
8-16	1.50E+01	1.14 E+03	5.28 E+03
16-24	1.50E+01	8.66 E+02	4.00 E+03
>24	1.50E+01	3.94 E+02	1.82 E+03

Note 9: During a main steam line break in containment or LOCA with temperature >170F, there is a potential to induce transient errors into the output of RE4596A and B during the peak rate of temperature change. Consult alternate indications. If the main steam line break is accompanied by core damage this error is insignificant (ref. 4, 5, 6, 7, 8).

Basis:

Generic

The radiation monitor reading corresponds to an instantaneous release of ALL reactor coolant mass into the containment, assuming that reactor coolant activity equals 300 µCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

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.

Plant-Specific

The containment high range monitors, RE 4596A & B., monitor the gamma dose rate resulting from a postulated loss of coolant accident (LOCA). RE 4596A & B are located inside containment. The detector range is approximately 1 to 1E8 R/hr (logarithmic scale) (ref. 1).

The Table F-2 values, column FC Loss represents, based on Calculation EP-EALCALC-DB-0701, the expected containment high range radiation monitor (RE 4596A & B) response based on a LOCA, for periods of 1, 2, 8, 16, 24 and 48 (>24) hours after shutdown with ~4.33% fuel failure (ref. 2).

When evaluating fission product barrier integrity values in Table F-2, time after shutdown should be confirmed with the Control Room as the time that the reactor is tripped, and reactor power is verified to be lowering on the Intermediate Range (ref. 3). If time after shutdown is less than one hour (or the reactor is still critical), the 0-1 hour after shutdown value should be chosen. This is conservative as it represents sufficient time for plant conditions to deteriorate to the point that core damage may occur, and the activity released from the RCS into the containment atmosphere to reach equilibrium mixing throughout containment.

During a main steam line break in containment or LOCA with temperature > 170ºF, there is a potential to induce transient errors (positive and negative) into the output of RE4596A & B during the peak rate of temperature change. Consult alternate indications. If the main steam line break or LOCA is accompanied by core damage this error is, however, insignificant (ref. 4).

DBNPS Basis Reference(s):

1. UFSAR Section 7.13.3.1 Containment High Radiation Monitors
2. EP-EALCALC-DB-0701 Containment Radiation Monitor Readings Following Clad Damage (FC2 and CT2 Potential Loss)
3. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
4. NRC Information Notice 97-45 Supplement 1 Environmental Qualification Deficiency for Cables and Containment Penetration Pigtails
5. Condition Report 09-53277, Containment High Range Rad Monitor Function and Current EALS
6. Condition Report 09-53278, Containment High Range Rad Monitor Function and NEI 99-01 EAL Submittal
7. Condition Report 07-31108, Potential for Thermally Induced Currents In Containment HRRM
8. Condition Report 09-55171, Containment High Range Rad Monitors Engineering Assistance Requested
9. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.A

Barrier: Fuel Clad

Category: C. CT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. Dose equivalent I-131 coolant activity > 300 µCi/gm

Basis:

Generic

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.

Plant-Specific

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The threshold dose equivalent I-131 concentration is well above that expected for iodine spikes and corresponds to about 4.33% fuel clad damage. When reactor coolant activity reaches this level the Fuel Clad barrier is considered lost (ref. 1).

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

DBNPS Basis Reference(s):

1. EP-EALCALC-DB-0701 Containment Radiation Monitor Readings Following Clad Damage (FC2 and CT2 Potential Loss)
2. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.B

Barrier: Fuel Clad

Category: C. CT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier

Basis:

Generic

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

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours 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 Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

DBNPS Basis Reference(s):

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

Barrier: Fuel Clad

Category: E. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier

Basis:

Generic

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

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours 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 Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

DBNPS 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 (SFAS) actuation required by **EITHER:**
   - UNISOLABLE RCS leakage
   - SG tube RUPTURE

Basis:

Generic

This threshold is based on an UNISOLABLE RCS leak of sufficient size to require an automatic or manual actuation of the Emergency Core Cooling System (ECCS). This condition clearly represents a loss of the RCS Barrier.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED. If a RUPTURED steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold A.1 will also be met.

Plant-Specific

ECCS (SFAS) actuation is caused by (ref. 1):

- Low RCS pressure
- High Containment pressure

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. 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 Makeup Pump is required (> 250 gpm) by **EITHER:**
   - UNISOLABLE RCS leakage
   - SG tube leakage

Basis:

Generic

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.

Plant-Specific

The Makeup and Purification System includes two centrifugal makeup pumps, which take suction from the Makeup Tank, and return cooled, purified reactor coolant to the RCS. One of the two makeup pumps handles normal charging flow. Makeup pump capacity of a single makeup injection line is ~250 gpm. A second makeup pump being required is indicative of a substantial RCS leak (ref. 1, 2).

DBNPS Basis Reference(s):

1. UFSAR Table 9.3-8 Makeup and Purification System Component Data
2. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
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. PTS requirements invoked (SR5)

Basis:

Generic

This condition indicates an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized).

Plant-Specific

With an extended overcooling, thermal shock becomes a concern. Pressurized Thermal Shock (PTS) limits must be invoked if the criteria specified in Specific Rule 5 are met. The RCS pressure must be controlled to ensure that PTS limits are not violated. This requires action on the part of the operator to control RCS pressure and temperature (ref. 1).

The "Potential Loss" threshold is defined by the PTS limits as specified in DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture Specific Rule 5 being invoked. This indicates an extreme challenge to the RCS barrier (ref. 1, 2).

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. Bases and Deviation Document for DB-OP-02000
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. Loss of **ALL** feedwater

   AND

   SG cooling is required

Basis:

Generic

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.

Plant-Specific

In combination with FC Potential Loss B.2, meeting this threshold results in a Site Area Emergency.

Loss of ALL feedwater cooling heat transfer capability when SG cooling is required indicates the ultimate heat sink function is under extreme challenge and that the RCS barrier is also challenged (ref. 1).

The phrase AND ”SG cooling required” precludes the need for classification for conditions in which RCS pressure is less than SG pressure. For large LOCA events inside the Containment, the SGs are moot because heat removal through the containment heat removal systems takes place. Therefore, SG cooling should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification (ref. 1, 2).

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. Bases and Deviation Document for DB-OP-02000
3. NEI 99-01 Inadequate Heat Removal RCS Loss 2.B

Barrier: Reactor Coolant System

Category: C. CT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

1. RE 4596A or B > Table F-2 column "RCS Loss" (Note 9)

Table F-2 Containment Radiation – R/hr (RE 4596A or B)			
Time After S/D (Hrs.)	**RCS Loss**	**FC Loss**	**CT Potential Loss**
0-1	1.50E+01	3.03E+03	1.40E+04
1-2	1.50E+01	2.56 E+03	1.18 E+04
2-8	1.50E+01	1.61 E+03	7.46 E+03
8-16	1.50E+01	1.14 E+03	5.28 E+03
16-24	1.50E+01	8.66 E+02	4.00 E+03
>24	1.50E+01	3.94 E+02	1.82 E+03

Note 9: During a main steam line break in containment or LOCA with temperature >170F, there is a potential to induce transient errors into the output of RE4596A and B during the peak rate of temperature change. Consult alternate indications. If the main steam line break is accompanied by core damage this error is insignificant (ref. 5, 6, 7, 8, 9).

Basis:

Generic

The radiation monitor reading corresponds to an instantaneous release of ALL reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold C.1 since it indicates a loss of the RCS Barrier only.

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

Plant-Specific

The containment high range monitors, RE 4596A & B., monitor the gamma dose rate resulting from a postulated loss of coolant accident (LOCA). RE 4596A & B are located inside containment. The detector range is approximately 1 to 1E8 R/hr (logarithmic scale) (ref. 1).

The Table F-2 values, column RCS Loss represents, based on Calculation EP-EALCALC-DB-0702, the expected containment high range radiation monitor (RE 4596A & B) response based on a LOCA, using the USAR maximum RCS activity (no core damage) (ref. 2, 3).

During a main steam line break in containment or LOCA with temperature > 170ºF, there is a potential to induce transient errors (positive and negative) into the output of RE4596A & B during the peak rate of temperature change. Consult alternate indications. If the main steam line break or LOCA is accompanied by core damage this error is, however, insignificant (ref. 5).

Since fuel cladding degradation and/or failures could also result in high Containment Area Radiation levels, a reading of >15 R/hr may be obtained without a physical loss of the RCS barrier. However, this threshold should be declared as being met if the Containment Area Radiation levels are >15 R/hr even if there are no other indications of a RCS leak or physical loss. In this case it would also be prudent to evaluate the fuel clad FISSION PRODUCT BARRIER THRESHOLDS.

DBNPS Basis Reference(s):

1. UFSAR Section 7.13.3.1 Containment High Radiation Monitors
2. EP-EALCALC-DB-0702 Containment Radiation Monitor Readings Following a LOCA (RC2 Loss)
3. USAR Table 15A-4 Maximum Fission Product Activity in Reactor Coolant
4. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
5. NRC Information Notice 97-45 Supplement 1 Environmental Qualification Deficiency for Cables and Containment Penetration Pigtails
6. Condition Report 09-53277, Containment High Range Rad Monitor Function and Current EALS
7. Condition Report 09-53278, Containment High Range Rad Monitor Function and NEI 99-01 EAL Submittal
8. Condition Report 07-31108, Potential for Thermally Induced Currents In Containment HRRM
9. Condition Report 09-55171, Containment High Range Rad Monitors Engineering Assistance Requested
10. NEI 99-01 CMT Radiation / RCS Activity RCS Loss 3.A

Barrier: Reactor Coolant System

Category: B. CT Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier

Basis:

Generic

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

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS Barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current SAFETY SYSTEM performance. The term "IMMINENT" refers to the 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 Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

DBNPS Basis Reference(s):

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

Barrier: Reactor Coolant System

Category: E. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier

Basis:

Generic

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

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS Barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current SAFETY SYSTEM performance. The term "IMMINENT" refers to the inability to reach final safety acceptance criteria before completing 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 Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

DBNPS 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

Basis:

Generic

This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for RCS Barrier Potential Loss A.1 and Loss A.1, respectively. This condition represents a bypass of the containment barrier.

FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably (part of the FAULTED definition) and the FAULTED steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition; the steam generator is still considered FAULTED for emergency classification purposes.

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU4 for the fuel clad barrier (i.e., RCS activity values) and IC SU5 for the RCS barrier (i.e., RCS leak rate values).

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, glad seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

The ECLs resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

	Affected SG is FAULTED Outside of Containment?	
P-to-S Leak Rate	**Yes**	**No**
Less than or equal to 10 gpm	No classification	No classification
Greater than 10 gpm for 15 minutes or longer	Unusual Event per SU5	Unusual Event per SU5
Requires operation of a standby charging (makeup) pump (*RCS Barrier Potential Loss*)	Site Area Emergency per FS1	Alert per FA1
Requires an automatic or manual ECCS (SI) actuation (*RCS Barrier Loss*)	Site Area Emergency per FS1	Alert per FA1

There is no Potential Loss threshold associated with RCS or SG Tube Leakage.

<u>Plant-Specific</u>

None

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. 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. Calculated Clad Temperature in **Region 3** or higher (DB-OP-02000 Figure 2)

   AND

   Restoration procedures **not** effective within 15 min. (Note 1)

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

Basis:

Generic

This condition represents an IMMINENT core melt sequence, which, if not corrected, could lead to vessel failure and an increased potential for containment failure. For this condition to occur there must already have been a loss of the RCS Barrier AND the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.

The restoration procedure is considered “effective” if core exit thermocouple readings are decreasing and/or if reactor vessel level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The Emergency Director should escalate the emergency classification level as soon as it is determined that the procedure(s) will not be effective.

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.

Plant-Specific

Indication of severe core cooling degradation is manifested by Calculated Clad Temperature in Region 3 (DB-OP-02000 Figure 2). Figure 2, Incore T/C Temperature vs. RCS Pressure for ICC, provides indication of how serious core conditions are based upon combinations of RCS pressure AND incore thermocouple temperatures. If the RCS P-T point is in Region 3, the cladding temperatures in the core may be 1400°F or higher. This is a very serious condition

and may lead to significant amounts of H2 production; core damage may be unavoidable as this represents a very serious inadequate core cooling condition (ref. 1, 2).

The function restoration procedures are those emergency operating procedures that address the recovery of core cooling functions. The procedure is considered effective if the clad temperature is decreasing or if RCS water level is increasing (ref. 1, 2).

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. Bases and Deviation Document for DB-OP-02000
3. NEI 99-01 Inadequate Heat Removal Containment Potential Loss 2.A

Barrier: Containment

Category: C. CT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: C. CT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

1. RE 4596A or B > Table F-2 column "CT Potential Loss" (Note 9)

Table F-2 Containment Radiation – R/hr (RE 4596A or B)			
Time After S/D (Hrs.)	**RCS Loss**	**FC Loss**	**CT Potential Loss**
0-1	1.50E+01	3.03E+03	1.40E+04
1-2	1.50E+01	2.56 E+03	1.18 E+04
2-8	1.50E+01	1.61 E+03	7.46 E+03
8-16	1.50E+01	1.14 E+03	5.28 E+03
16-24	1.50E+01	8.66 E+02	4.00 E+03
>24	1.50E+01	3.94 E+02	1.82 E+03

Note 9: During a main steam line break in containment or LOCA with temperature >170F, there is a potential to induce transient errors into the output of RE4596A and B during the peak rate of temperature change. Consult alternate indications. If the main steam line break is accompanied by core damage, this error is insignificant.

Basis:

Generic

The radiation monitor reading corresponds to an instantaneous release of ALL reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment, which would then escalate the ECL to a General Emergency.

Plant-Specific

The containment high range monitors, RE 4596A & B., monitor the gamma dose rate resulting from a postulated loss of coolant accident (LOCA). RE 4596A & B are located inside containment. The detector range is approximately 1 to 1E8 R/hr (logarithmic scale) (ref. 1).

The Table F-2 values, column CT Potential Loss represents, based on Calculation EP-EALCALC-DB-0701, the expected containment high range radiation monitor (RE 4596A & B) response based on a LOCA, for periods of 1, 2, 8, 16, 24 and 48 (>24) hours after shutdown with ~20% fuel failure (ref. 2).

When evaluating fission product barrier integrity values in Table F-2, time after shutdown should be confirmed with the Control Room as the time that the reactor is tripped, and reactor power is verified to be lowering on the Intermediate Range (ref. 3). If time after shutdown is less than one hour (or the reactor is still critical), the 0-1 hour after shutdown value should be chosen. This is conservative as it represents sufficient time for plant conditions to deteriorate to the point that core damage may occur, and the activity released from the RCS into the containment atmosphere to reach equilibrium mixing throughout containment.

During a main steam line break in containment or LOCA with temperature > 170ºF, there is a potential to induce transient errors (positive and negative) into the output of RE4596A & B during the peak rate of temperature change. Consult alternate indications. If the main steam line break or LOCA is accompanied by core damage this error is, however, insignificant (ref. 4).

DBNPS Basis Reference(s):

1. UFSAR Section 7.13.3.1 Containment High Radiation Monitors
2. EP-EALCALC-DB-0701 Containment Radiation Monitor Readings Following Clad Damage (FC2 and CT2 Potential Loss)
3. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
4. NRC Information Notice 97-45 Supplement 1 Environmental Qualification Deficiency for Cables and Containment Penetration Pigtails
5. NEI 99-01 CMT Radiation / RCS Activity Containment Potential Loss 3.A

Barrier: Containment

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. Containment isolation is required

   AND EITHER:

   - Containment integrity has been lost based on Emergency Director judgment
   - UNISOLABLE pathway from Containment to the environment exists

Basis:

Generic

These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both bulleted thresholds.

First Threshold – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the 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.).

Refer to the middle piping run of Figure 1. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

Another example would be a loss or potential loss of the RCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment

leakage through various penetrations or system components. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

Second Threshold – Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment atmosphere to the environment. As used here, the term “environment” includes the atmosphere of a room or area, outside the containment, that may, in turn, communicate with the outside-the-plant atmosphere (e.g., through discharge of a ventilation system or atmospheric leakage). Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.

Refer to the top piping run of Figure 1. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Leakage between two interfacing liquid systems, by itself, does not meet this threshold.

Refer to the bottom piping run of Figure 1. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump developed a leak that allowed steam/water to enter the Auxiliary Building, then second threshold would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause the first threshold to be met as well.

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

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold A.1.

Plant-Specific

None

DBNPS Basis Reference(s):

1. NEI 99-01 CMT Integrity or Bypass Containment Loss 4.A

Barrier: Containment

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

2. Indications of RCS leakage outside of Containment

Basis:

Generic

Containment sump level, 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 level, temperature, pressure, flow and/or radiation level readings outside of the containment may indicate that the RCS mass is being lost outside of containment.

Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of RCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside of containment may not increase significantly; however, other unexpected changes in sump levels, area temperatures or pressures, flow rates, etc. should be sufficient to determine if RCS mass is being lost outside of the containment.

Refer to the middle piping run of Figure 1. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause threshold D.1 to be met as well.

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.

Plant-Specific

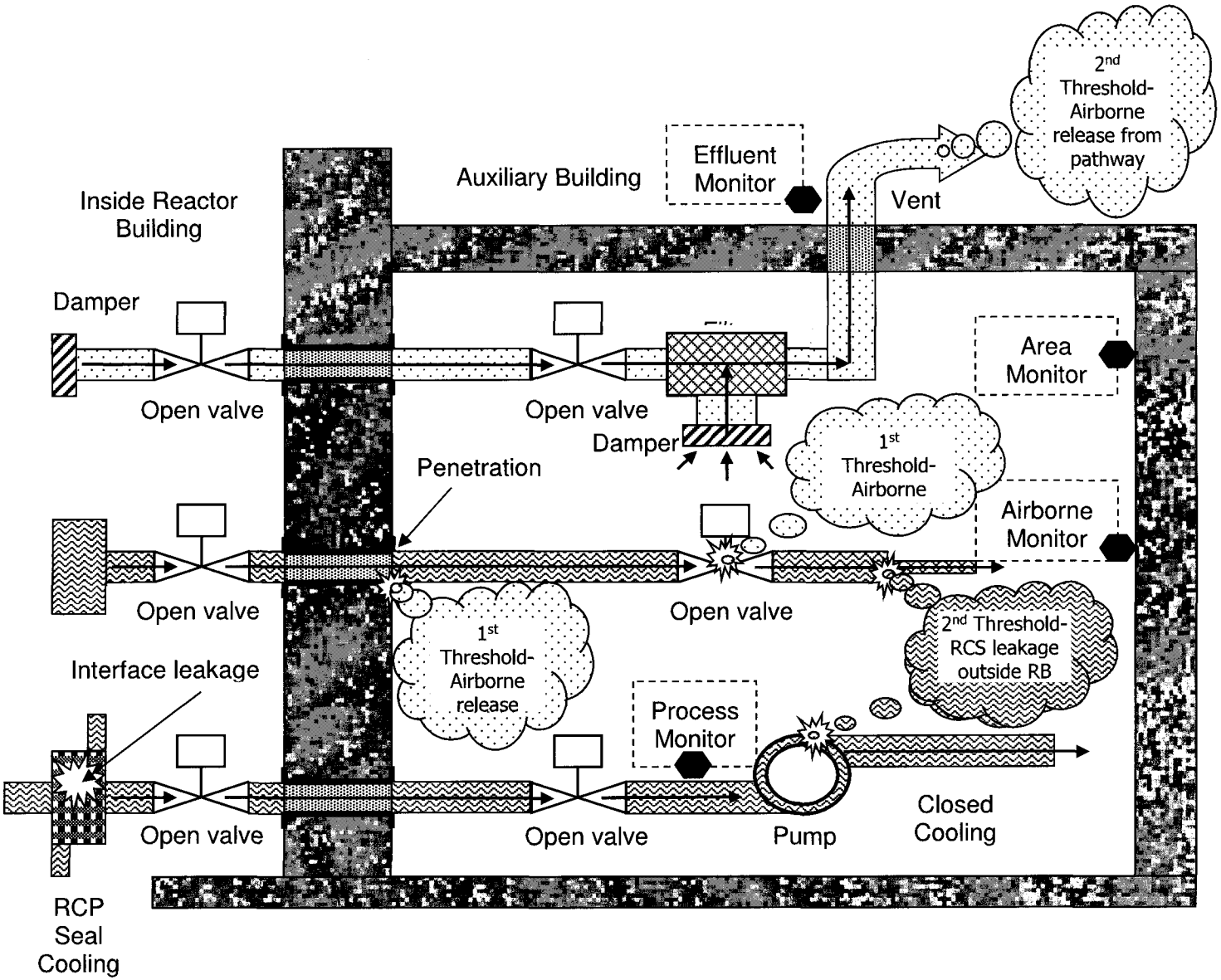
Potential RCS leak pathways outside containment include (ref. 1, 2):

- Decay Heat Removal
- ECCS (Safety Injection)
- Makeup and Purification
- RC pump seals
- RCS sample lines
- RCS drain lines

DBNPS Basis Reference(s):

1. UFSAR Section 5.2.4.7 Leakage Identification
2. DB-OP-02522 Small RCS Leaks
3. NEI 99-01 CMT Integrity or Bypass Containment Loss

Figure 1: Containment Integrity or Bypass Examples



Barrier: Containment

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

1. Containment pressure > 50.4 psia

Basis:

Generic

If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the RCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a Site Area Emergency and General Emergency since there is now a potential to lose the third barrier.

Plant-Specific

50.4 psia (36 psig + elevation adjusted atmospheric pressure of 14.4 psia) is based on the containment design pressure (ref.1).

DBNPS Basis Reference(s):

1. UFSAR Section 3.8.2.1 Containment Vessel
2. 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%

Basis:

Generic

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

Plant-Specific

Following a design basis accident, hydrogen gas may be generated inside the containment by reactions such as zirconium metal with water, corrosion of materials of construction and radiolysis of aqueous solution in the core and sump (ref. 1).

The Containment Hydrogen Monitoring System is used to monitor the hydrogen concentration inside containment after a severe accident involving core damage. The containment hydrogen monitors (AI 5027 & AI 5028) are not required to be operated in the continuous mode, however, the system is required to be started up 30 minutes after containment sprays have been initiated (ref. 2).

The lower limit for the occurrence of an in-containment hydrogen burn is approximately 4% (ref. 1).

To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers must have occurred. With the Potential Loss of the containment barrier, the threshold hydrogen concentration, therefore, will likely warrant declaration of a General Emergency.

DBNPS Basis Reference(s):

1. DBSAMG-TBD Davis-Besse Severe Accident Management Guidelines Technical Bases Document
2. UFSAR Section 7.13.3.4 Containment Hydrogen Monitors
3. 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 > 40 psia with < one full train of containment cooling, Table F-3, operating per design for ≥ 15 min. (Note 1)

Table F-3 Containment Cooling Full Train	
CT Spray Pumps	**CT Cooling Fans**
2	0
1	1
0	2

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

Basis:

Generic

This threshold describes a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems (e.g., containment sprays, ice condenser fans, etc., but not including containment venting strategies) are either lost or performing in a degraded manner.

Plant-Specific

The combination of Containment spray pumps and Containment cooling fan units considered to be a full train of containment cooling operating per design is shown in Table F-3 (ref. 1).

SFAS 2 actuation automatically initiates Containment Air Coolers upon exceeding the Containment pressure high setpoint of 18.7 psia or low RCS pressure of 1600 psig. SFAS Level 4 actuation automatically initiates Containment Spray upon exceeding the Containment pressure high-high setpoint of 40 (nominal) psia (ref. 2, 3).

DBNPS Basis Reference(s):

1. UFSAR Section 6.2.2. Containment Vessel Heat Removal Systems
2. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
3. DBNPS Technical Specifications Table 3.3.5-1 Safety Features Actuation System Instrumentation
4. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.C

Barrier: Containment

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier

Basis:

Generic

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

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours 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 Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

DBNPS Basis Reference(s):

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

Barrier: Containment

Category: F. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier

Basis:

Generic

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

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours 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 Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

DBNPS 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.*

DBNPS Table R-2 and H-2 Bases

NEI 99-01 Rev 06 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.

DB-OP-06902, Power Operations Rev 46 was reviewed to determine what actions are "necessary" to maintain power operations. It was determined that over reasonable periods of time (days vice years) there are no actions outside the Control Room that are required to be performed to maintain normal operations. Eventually, a shutdown would be required if Technical Specification surveillance testing was not completed and you complied with the associated LCO's 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 that an operator may be dispatched in order perform a normal plant cooldown and shutdown. The review was completed using the following procedures as the controlling documents:

DB-OP-06902, Power Operations R46

DB-OP-06903, Plant Cooldown R47

DB-OP-02504, Rapid Shutdown R20

In addition, DB-OP-06012 was reviewed to ensure the Decay Heat Removal System is aligned.

At Davis-Besse, RCS Cooldown starts once both Steam Generators reach Low Level Limits during a power reduction (approximately 30% power). As a result, this review started with DB-OP-06902, Power Operations, Section 8, Turbine and Reactor Shutdown and then transitioned to DB-OP-06903, Plant Cooldown. Each step in the controlling procedures was evaluated to determine if the action was performed in the Control Room or in the plant. In-plant actions were evaluated and a determination was made whether or not the actions, if not performed, would prevent achieving cold shutdown. The following generic assumptions were applied:

- Steps involving optional degasing of the RCS were not included since degasing the RCS is not required to reach cold shutdown.
- Steps involving supplying Auxiliary Steam were not included since AFW and AVVs can be used to reach cold shutdown if Condenser vacuum is lost.
- Steps involving Main Feedwater Pumps were not included since AFW and AVVs can be used to reach cold shutdown if Main Feedwater is not available.
- Travel paths to the locations where the equipment is operated are not part of the determination, only the rooms where the equipment is actually operated are considered

as the affected room. Travel paths were not included because most locations can be reached via alternate travel paths if required due to a localized issue.

- No assumptions made about which LPI Train is aligned for DHR Operation. Locations could be reduced by preselecting one train (typically Train 2) to provide this function. It is assumed that both trains are in a Standby LPI mode at the start of the event.

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 in EAL Tables R-2 and H-2.

The control room was not included in Table H-2 evaluation because the control room is governed by H6 series for Control Room Evacuation.

UFSAR Section 6.4.2 Toxic Gas Protection Provisions states that no toxic or explosive materials are stored in volumes or locations, which pose a control room habitability hazard that exceeds emergency system capabilities.

EAL RA3.1 addresses control room habitability relative to area radiation levels.

In Plant Task - Procedure and Step	Step Action	If action not performed, does this prevents cooldown shutdown?	Building	Elevation	Room	Mode
06902 8.2.2	Align CWRT to RCS Makeup.	No – Inventory and Boration requirements can be met from Control Room using BWST and BAATs.	N/A	N/A	N/A	N/A
06902 8.2.3	Turbine Overspeed trip testing if required.	No – This testing is not required. Testing could be performed on subsequent restart.	N/A	N/A	N/A	N/A
06902 8.3.9 Bullet	Set condenser pressure to 1.5 to 2.5 inches HgA using PCV1061.	No – This action is not required to perform Shutdown – Cooldown.	N/A	N/A	N/A	N/A
06902 8.3.9 Bullet	Attachment 8, Drain and Steam Trap Alignment for Shutdown	No – This only affects efficiency or improves moisture removal.	N/A	N/A	N/A	N/A
06902 8.3.9 Bullet	Fill and Vent the SG Blowdown lines,	No – This action facilitates SG Fill, Soak, and Drains, but is not required to complete Shutdown – Cooldown.	N/A	N/A	N/A	N/A
06902 8.3.9 Bullet	Place both Instrument Air dryers in service in parallel,	No - Step improves response of air system, but is not required.	N/A	N/A	N/A	N/A
06902 8.3.9 Bullet	MDFP is in Standby in the Main Feedwater mode and warm up is complete.	No – MDFP could be used in AFW Mode to reach Cold Shutdown.	N/A	N/A	N/A	N/A
06902 8.3.9 Bullet	Place the Auxiliary Boiler in service.	No – Aux Steam will continue to be supplied from Main Steam. If vacuum and therefore condenser is lost, steam can be dumped via AVV's.	N/A	N/A	N/A	N/A
6902 8.3.9 Bullet	Transfer Auxiliary Steam Loads from the Main Steam Reducing Station to the Auxiliary Boiler.	No – Aux Steam will continue to be supplied from Main Steam. If vacuum and therefore condenser is lost, steam can be dumped via AVV's.	N/A	N/A	N/A	N/A

In Plant Task - Procedure and Step	Step Action	If action not performed, does this prevents cooldown shutdown?	Building	Elevation	Room	Mode
6902 8.3.10 Bullet	Perform Attachment 10, MDFP Operation.	No – MDFP could be used in AFW Mode to reach Cold Shutdown.	N/A	N/A	N/A	N/A
6902 8.3.10 Bullet	Closed the actuator cylinder equalizing valve for MS4531/MS4532	No – This action facilitates SG Fill, Soak, and Drains, but is not required to complete Shutdown – Cooldown.	N/A	N/A	N/A	N/A
6902 8.3.10 Bullet	Place the SG Blowdown Lines in service.	No – This action facilitates SG Fill, Soak, and Drains, but is not required to complete Shutdown – Cooldown.	N/A	N/A	N/A	N/A
6902 8.3.16	IF Main Turbine Overspeed Testing is required,	No – This testing is not required. Testing could be performed on subsequent restart.	N/A	N/A	N/A	N/A
6902 8.3.17	Perform Attachment 13, Turbine Shutdown.	No – This action completes Turbine Shutdown, but is not required to meet Cold Shutdown.	N/A	N/A	N/A	N/A
6902 8.3.30	Open the air isolation valve that was closed in Attachment 10, MDFP Operation:	No – Since performance of Attachment 10 is not required, neither is restoration.	N/A	N/A	N/A	N/A
6902 8.3.31	Complete the shutdown of the running MFPT to Turning Gear operation.	No – desired to complete MFPT shutdown, but will not prevent reaching cold shutdown.	N/A	N/A	N/A	N/A
6902 8.3.32	Perform Attachment 8, Drain and Steam Trap Alignment for Shutdown, step 3.0.	No – This only affects efficiency or improves moisture removal.	N/A	N/A	N/A	N/A
6902 8.3.38	Transition to DB-OP-06903, Plant Cooldown	No – This is a procedure routing step.	N/A	N/A	N/A	N/A
6903 3.3.10	Perform the following to read Reactor Vessel Head O-Ring pressure.	No – Action not required to reach cold shutdown.	N/A	N/A	N/A	N/A

In Plant Task - Procedure and Step	Step Action	If action not performed, does this prevents cooldown shutdown?	Building	Elevation	Room	Mode
6903 3.12.1	Verify the requirements pertaining to containment closure control and protected equipment are initiated.	No – Action not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.12.2	Initiate Attachment 13, Preparation of MU Filter 1 for Hydrogen Peroxide Addition.	No – Action not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.13	Direct I & C Department to perform Attachment 8, Radiation Monitor Preparations for Plant Shutdown	No – Action not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.16	IF the RCS will be opened to atmosphere, THEN begin to reduce the RCS H2 concentration to less than 15 cc/kg.	No – Opening the RCS is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.17	IF the Circulating Water System is to be drained, THEN notify Chemistry to chemically shock it prior to bypassing the Cooling Tower.	No – Draining Circ Water is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.18	Remove TPCW pumps and heat exchangers from service, REFER TO DB-OP-06263, Turbine Plant Cooling Water	No – Controlling temperature and Shutdown of TPCW is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.21	Verify one of the Clean Waste Receiver Tanks is aligned…	No – The BWST and the BAATs can be used from the Control Room for RCS inventory and Boration.	N/A	N/A	N/A	N/A
6903 3.25	Begin making preparations to Start-up Containment Vessel Purge.	No – Placing CTMT Purge in service is not required to reach cold shutdown.	N/A	N/A	N/A	N/A

In Plant Task - Procedure and Step	Step Action	If action not performed, does this prevents cooldown shutdown?	Building	Elevation	Room	Mode
6903 3.25.2.a/d.3	Place/remove a second seal return cooler in/from service.	No – Controlling a second seal return cooler not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.35.2	Begin performing Attachment 5 SG Fill, Soak, and Drain.	No – This action facilitates SG Fill, Soak, and Drains, but is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.47	Shutdown one RCP.	No – The controlling procedure provides direction when CTMT is not accessible. That direction would be used.	N/A	N/A	N/A	N/A
6903 3.53	Isolate Core Flood Tank 1 perform the following:	Yes – Action is required to reduce RCS Pressure. CFT 1 Outlet Valve power restored at E11B to allow closure.	Aux	585	Rm 304 – Corridor Outside #3 MPR	1, 2, 3
6903 3.53	Isolate Core Flood Tank 2 perform the following:	Yes – Action is required to reduce RCS Pressure. CFT 2 Outlet Valve power restored at F11A to allow closure.	Aux	603	Rm 427 - #2 Electrical Penetration Room	1, 2, 3
06903 3.63	WHEN RCS pressure is between 450 psig and 425 psig THEN establish a 0/2 or 2/0 RCP combination.	No – The controlling procedure provides direction when CTMT is not accessible. That direction would be used.	N/A	N/A	N/A	N/A
6903 3.64	Control Letdown Flow - Open or throttle MU 83	No – This action aids in RCS cleanup, which is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.64.1	IF its necessary to reduce the total makeup flow to the RCS by throttling MU 58A, NORMAL MAKEUP FLOW FE MU58 SOURCE/NEEDLE,	No – This action allow lower Makeup Flow, which would only be required in a very slow cooldown. This action is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.66	WHEN Feedwater flow reduces to the point where level control using the Feedwater Startup Valves is difficult, Throttle FW 139 and/or FW44.	No – This action would reduce FW flow via the MFW header, which would only be required in a very slow cooldown. This action is not required to reach cold shutdown.	N/A	N/A	N/A	N/A

In Plant Task - Procedure and Step	Step Action	If action not performed, does this prevents cooldown shutdown?	Building	Elevation	Room	Mode
6903 3.68	Start actions to reduce feedwater temperature to 200F.	No – This action will allow cooldown to proceed quicker if on Main Feedwater, but is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 4.17.1	To prevent a transfer of water from the RCS to the BWST, close DH10.	Yes – Action is required to align LPI Train 1 for DHR Operations.	Aux	565	Room 236 #2 Mechanical Penetration Room	1, 2, 3
6903 4.17.1	To prevent a transfer of water from the RCS to the BWST, close DH26.	Yes – Action is required to align LPI Train 2 for DHR Operations.	Aux	565	Room 236 #2 Mechanical Penetration Room	1, 2, 3
6903 4.17.2	Close breaker BF 1130 in F11A, for DH 11	Yes – Action is required to align either LPI Train 1 or 2 for DHR Operations.	Aux	603	Rm 427 - #2 Electrical Penetration Room	1, 2, 3
6903 4.17.2	Close breaker BE 1183 in E11B, for DH 12.	Yes – Action is required to align either LPI Train 1 or 2 for DHR Operations.	Aux	585	Rm 304 – Corridor Outside #3 MPR	1, 2, 3
6903 4.17.8.d	If DH12 does not open (which is generally does not), Install jumper.	Yes – Action is required to align either LPI Train 1 or 2 for DHR Operations.	Aux	585	Rm 304 – Corridor Outside #3 MPR	1, 2, 3
6903 4.20	Begin reducing Deaerator and Feedwater temperature to less than 120F.	No – This action will allow cooldown to proceed quicker if on Main Feedwater, but is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 4.21	WHEN RCS temperature is less than 280ºF, THEN disable HPI by racking out breakers.	No – This action will prevent low temp-high pressure conditions in RCS, but is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 4.22.3	IF MSIVs are to be stroked or pinned, THEN stroke test MS100 and MS101.	No – Action is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 4.25	Place a LPI Train in service as a Decay Heat Removal Train,	Yes – This action is performed per DB-OP-06012. However, since numerous actions are required each action in that procedure was assessed separately.	N/A	N/A	N/A	N/A

In Plant Task - Procedure and Step	Step Action	If action not performed, does this prevents cooldown shutdown?	Building	Elevation	Room	Mode
6903 4.28	Remove the Auxiliary Feedwater System from service.	No – Action is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 4.30	IF SG 1 level control becomes difficult due to excess flow through SP7B, adjust FW161/139.	No – Action is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 4.32	IF SG 2 level control becomes difficult due to excess flow through SP7A, adjust FW162/44.	No – Action is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 4.34	WHEN RCS temperature is less than 200ºF,	No – Cold shutdown has been reached. End of review using DB-OP-06903.	N/A	N/A	N/A	N/A
6012 3.5.5	Verify the CLOSE power fuses for AC 112, DECAY HT PUMP 1-1, are removed.	No – Just prevents inadvertent start of LPI pump 1 while transferring suction. SFAS is already blocked.	N/A	N/A	N/A	N/A
6012 3.5.6	Verify BE 1187 (E11E), MV DH64 LPI-HPI CROSS CONN ISO VLV 1, is open.	No – Just prevents possible valve motor overload if stroked with RCS Suction Source.	N/A	N/A	N/A	N/A
6012 3.5.10	Open BE 1121 (E11A), MV 2733 DH PUMP 1 SUCT VLV FRM BWST.	No – Just prevents inadvertent transfer of BWST inventory to RCS if the valve opened. SFAS is already blocked.	N/A	N/A	N/A	N/A
6012 3.5.13	Close DH10*, DH PUMP 1 MINIMUM COOLDOWN ISOLATION.	Yes – This action is required to align LPI Train 1 or 2 for DHR Operations.	Aux	565	Room 236 #2 Mechanical Penetration Room	1, 2, 3
6012 3.5.17	Open BE 1126 (E11D), MV 1517 DH NORM SUCT LINE 1 ISO VLV.	No - Just prevent inadvertent loss of DHR Train 1 suction from RCS is valve is stroked closed.	N/A	N/A	N/A	N/A
6012 3.6.5	Verify the CLOSE power fuses for AD 112, DECAY HT PUMP 1-2, are removed.	No – Just prevents inadvertent start of LPI pump 2 while transferring suction. SFAS is already blocked.	N/A	N/A	N/A	N/A

In Plant Task - Procedure and Step	Step Action	If action not performed, does this prevents cooldown shutdown?	Building	Elevation	Room	Mode
6012 3.6.6	Verify BE 1195 (F11E), MV DH63 LPI-HPI CROSS CONN ISO VLV 2, is open.	No – Just prevents possible valve motor overload if stroked with RCS Suction Source.	N/A	N/A	N/A	N/A
6012 3.6.11	Open BF 1134 (F11C), MV 2734 DH PMP 2 SUCT VLV FRM BWST.	No – Just prevents inadvertent transfer of BWST inventory to RCS if the valve opened. SFAS is already blocked.	N/A	N/A	N/A	N/A
6012 3.6.14	Close DH10*, DH PUMP 1 MINIMUM COOLDOWN ISOLATION.	Yes – Action is required to align LPI Train 1 or 2 for DHR Operations.	Aux	565	Room 236 #2 Mechanical Penetration Room	1, 2, 3
6012 3.6.19	Open BF1129 (F11C), MV 1518 DH NORM SUCT LINE 2 ISO VLV.	No - Prevents inadvertent loss of DHR Train 2 suction from RCS is valve is stroked closed.	N/A	N/A	N/A	N/A
6012 3.6.27	Verify the CLOSE power fuses for AD 112.	No – If not initially removed, then reinstalling will not be required.	N/A	N/A	N/A	N/A
6012 3.7.8.a	Station an operator at DH Pump 1.	No – Operator only stationed for monitoring function. Action is not required to reach Cold Shutdown.	N/A	N/A	N/A	N/A
6012 3.7.8.d	Verify DH59, DH PUMP 1 DISCHARGE SAMPLE ISOL, is open.	No – Opening this valve provides the capability to sample RCS inventory from the DHR system.	N/A	N/A	N/A	N/A
6012 3.8.8.a	Station an operator at DH Pump 2.	No – Operator only stationed for monitoring function. Action is not required to reach Cold Shutdown.	N/A	N/A	N/A	N/A
6012 3.8.8.d	Verify DH60, DH PUMP 2 DISCHARGE SAMPLE ISOL, is open.	No – Opening this valve provides the capability to sample RCS inventory from the DHR system.	N/A	N/A	N/A	N/A

Note: The information in the above table is included for historical reference information only and based upon the procedure revision numbers referenced in the DBNPS Table R-2 and H-2 Bases summary.
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Table R-2 & H-2 Results

Table R-2 & H-2 Safe Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Aux Bldg. 565' ele. Room 236 #2 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 585' ele. Room 304 corridor outside #3 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 603' ele. Room 427 - #2 Electrical Penetration Room	1, 2, 3

Evaluation of Proposed License Amendment

Attachment 2

Emergency Action Level (EAL) Bases Document (Redline Version)

(292 Pages Follow)

EMERGENCY ACTION LEVEL TECHNICAL BASES DOCUMENT

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 Davis-Besse Nuclear Power Station (DBNPS). It should be used to facilitate review of the DBNPS EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of RA-EP-01500 Emergency Classification, may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Director in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training and for explaining event classifications to off-site officials.

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

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

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 DBNPS Emergency Plan (ref. 4.1.13).

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 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," (ref. 4.1.1), DBNPS 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 (RC): The RC 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 (CT): 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 RC 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 DBNPS EAL scheme includes the following features:

- Division of the EAL set into three broad groups:

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

    - EALs applicable only under hot operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Hot Standby, Startup, or Power Operation mode.

    - EALs applicable only under cold operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling or Defueled mode.

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

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

Recognition category and subcategory titles were selected to represent conditions that are operationally significant to the EAL-user. The DBNPS EAL recognition categories align to and represent the NEI 99-01 Revision 6 “Recognition Categories.” Subcategories are used in the DBNPS scheme to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The DBNPS EAL categories and subcategories are listed below.

EAL Groups, Recognition Categories and Subcategories

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

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

2.5 Technical Bases Information

EAL technical bases are provided in Attachment 1 for each EAL according to EAL group (ANY, Hot, Cold), EAL category (R, C, H, S, F and E) 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 recognition category as described in Section 2.4 (H, R, E, C, S or F)

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

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

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

4. Fourth character (number): The numerical sequence of the EAL within the EAL subcategory. If the subcategory has only one EAL, it is given the number one (1).

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 ALL. (See Section 2.6 for operating mode definitions)

Basis:

A Generic basis section that provides a description of the rationale for the EAL as provided in NEI 99-01 Rev. 6. This is followed by a Plant-Specific basis section that provides DBNPS-relevant information concerning the EAL.

DBNPS Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.6 Operating Mode Applicability (ref. 4.1.8)

Mode	Reactivity Condition (K_{eff})	% Rated Power*	Avg. Reactor Coolant Temperature (ºF)
1) Power Operation	≥ 0.99	> 5%	N/A
2) Startup	≥ 0.99	≤ 5%	N/A
3) Hot Standby	< 0.99	N/A	≥ 280
4) Hot Shutdown	< 0.99	N/A	280 > T_{avg} > 200
5) Cold Shutdown	< 0.99	N/A	≤ 200
6) Refueling	One or more vessel head closure bolts less than fully tensioned.		
D) Defueled	ALL reactor fuel removed from reactor pressure vessel (full core offload during refueling or extended outage).		

Refer to Section 3.3.2 for guidance on event caused mode changes.

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Emergency Director must consider ALL information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes, and the technical basis information. In the Recognition Category F matrices, EALs are based on loss or potential loss of FISSION PRODUCT BARRIER THRESHOLDS.

3.1.1 Classification Timeliness

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, "Interim Staff Guidance, Emergency Planning for Nuclear Power Plants" (ref. 4.1.11).

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.

An indication, report, or condition is considered 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. The validation of indications should be completed in a manner that supports timely emergency declaration.

3.1.3 IMMINENT Conditions

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

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) (ref. 4.1.11).

3.1.6 Emergency Director 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 Emergency Director with the ability to classify events and conditions based upon judgment using EALs that are consistent with the Emergency Classification Level (ECL) definitions (refer to Category H). The Emergency Director will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated in the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

3.1.7 Emergency Action Levels with Embedded Time Requirements

Some EALs have embedded time requirements. Declaration must be made as soon as the Emergency Director recognizes that the conditions will not be successfully resolved within 15 minutes. Therefore, for EALs with time-embedded requirements the time for emergency declaration starts with the initial alarm or indication of the event, not after the embedded time has elapsed.

For EALs with longer embedded time requirements, the 15-minute clock for declaration begins with recognition that the assigned time limit will be exceeded. For example, SG1.1 - “Restoration of at least one essential bus in < 4 hours is **not** likely.” If 20 minutes after loss of the C-1 bus and the D-1 bus it becomes apparent that restoration of either bus will not likely occur within 4 hours the Emergency Director must immediately declare the General Emergency.

3.2 Classification Methodology

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

met, the emergency classification process “clock” starts, and the ECL must be declared in accordance with plant procedures no later than fifteen minutes after the process “clock” started.

When assessing an EAL that specifies a time duration for the off-normal condition, the “clock” for the EAL time duration runs concurrently with the emergency classification process “clock.” For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01 (ref. 4.1.11).

3.2.1 Classification of Multiple Events and Conditions

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

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

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

3.2.3 Classification of IMMINENT Conditions

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

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 be terminated.

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

3.2.5 Classification of Short-Lived Events

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

3.2.6 Classification of Transient Conditions

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

<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.

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

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. 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 ML13091A209

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

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

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

4.1.5 10 § CFR 50.73 License Event Report System

4.1.6 DBNPS ODCM

4.1.7 DBNPS UFSAR Figure 1.2-12 Site Plan

4.1.8 Technical Specifications Table 1.1-1 Modes

4.1.9 DB-OP-06904 Shutdown Operations

4.1.10 NG-QS-00121 DBNPS Writer's Guide

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

4.1.12 Site CSAR NUH-003

4.1.13 DBNPS Emergency Plan

4.1.14 DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tube Rupture

4.2 Implementing

4.2.1 RA-EP-01500 Emergency Classification

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

4.2.3 DBNPS EAL Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

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

Alert

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

Confinement Boundary

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the DBNPS Dry Fuel Storage Facility, CONFINEMENT BOUNDARY is defined as the Dry Shielded Canister (DSC) (ref. 4.1.12).

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~~ existing plant conditions (ref. 4.1.9).

Emergency Action Level (EAL)

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

Emergency Classification Level (ECL)

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

- ~~Notification of~~ Unusual Event (~~NO~~UE)
- Alert
- Site Area Emergency (SAE)
- General Emergency (GE)

EPA PAGs

Environment Protection Agency Protective Action Guidelines. The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs requires DBNPS 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 may require a post-event inspection to determine if the attributes of an EXPLOSION are present.

Faulted

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

Fire

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

Fission Product Barrier Threshold

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

Flooding

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

General Emergency

Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Hostage

A person(s) held as leverage against the station to ensure that demands will be met by the station.

Hostile Action

An act toward DBNPS 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 DBNPS. Non-terrorism-based EALs should be

used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

Hostile Force

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

Imminent

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

Impede(d)

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

Independent Spent Fuel Storage Installation (ISFSI)

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

Initiating Condition (IC)

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

Maintain

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

~~**Normal Levels**~~

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

Owner Controlled Area

The property associated with the station and owned by the company. Access is normally limited to persons entering for official business (ref. 4.1.13).

Projectile

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

Protected Area

An area that normally encompasses all controlled areas within the security protected area fence.

RCS Intact

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

Refueling Pathway

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

Restore

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

Ruptured

The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection (ref. 4.1.14).

Safety System

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

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

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

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

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

Security Condition

ANY Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

Site Area Emergency

Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the SITE BOUNDARY.

Site Boundary

Area as depicted in UFSAR Figure 1.2-12 Site Plan (ref. 4.1.7). The SITE BOUNDARY is defined at a minimum exclusion distance of 0.75 miles. This is the nearest distance from potential release points at which protective actions would be required for members of the public.

Unisolable

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

Unplanned

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

Unusual Event

Events are in progress or have occurred which indicate a potential degradation 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.

Valid

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

Visible Damage

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

5.2 Abbreviations/Acronyms

°F ..... Degrees Fahrenheit
° ..... Degrees
AC ..... Alternating Current
AOP ..... Abnormal Operating Procedure
ATWS ..... Anticipated Transient Without Scram
B&W ..... Babcock & Wilcox
BWST ..... Borated Water Storage Tank
DBNPS ..... Davis-Besse Nuclear Power Station
CCW ..... Component Cooling Water
CDE ..... Committed Dose Equivalent
CFR ..... Code of Federal Regulations
DBA ..... Design Basis Accident
DC ..... Direct Current
DFSF ..... Dry Fuel Storage Facility
DHR ..... Decay Heat Removal
DSC ..... Dry Shielded Canister
EAL ..... Emergency Action Level
ECCS ..... Emergency Core Cooling System
ECL ..... Emergency Classification Level
EOF ..... Emergency Operations Facility
EOP ..... Emergency Operating Procedure
EPA ..... Environmental Protection Agency
EPIP ..... Emergency Plan Implementing Procedure
ERG ..... Emergency Response Guideline
ESF ..... Engineered Safety Feature
FAA ..... Federal Aviation Administration
FBI ..... Federal Bureau of Investigation
FEMA ..... Federal Emergency Management Agency
FSAR ..... Final Safety Analysis Report
GE ..... General Emergency
IC ..... Initiating Condition
IPEEE ..... Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI ..... Independent Spent Fuel Storage Installation

K_{eff} ..... Effective Neutron Multiplication Factor
LCO ..... Limiting Condition of Operation
LER ..... Licensee Event Report
LOCA ..... Loss of Coolant Accident
MSIV ..... Main Steam Isolation Valve
MSL ..... Main Steam Line
MSSV ..... Main Steam Safety Valve
mR, mRem, mrem, mREM ..... milli-Roentgen Equivalent Man
MU-HPI ..... Makeup and High Pressure Injection
MW ..... Megawatt
NEI ..... Nuclear Energy Institute
NESP ..... National Environmental Studies Project
NRC ..... Nuclear Regulatory Commission
NSSS ..... Nuclear Steam Supply System
NORAD ..... North American Aerospace Defense Command
OBE ..... Operating Basis Earthquake
OCA ..... Owner Controlled Area
ODCM ..... Off-site Dose Calculation Manual
ORO ..... Offsite Response Organization
PA ..... Protected Area
PAG ..... Protective Action Guideline
PORV ..... Power Operated Relief Valve
PRA/PSA ..... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR ..... Pressurized Water Reactor
PSIG ..... Pounds per Square Inch Gauge
R ..... Roentgen
RCDT ..... Reactor Coolant Drain Tank
RCS ..... Reactor Coolant System
Rem, rem, REM ..... Roentgen Equivalent Man
RPS ..... Reactor Protection System
RV ..... Reactor Vessel
SAR ..... Safety Analysis Report
SBO ..... Station Blackout
SBODG ..... Station Blackout Diesel Generator

SCBA ..... Self-Contained Breathing Apparatus
SG ..... Steam Generator
SI ..... Safety Injection
SPDS ..... Safety Parameter Display System
SRO ..... Senior Reactor Operator
TEDE ..... Total Effective Dose Equivalent
TOAF ..... Top of Active Fuel
TSC ..... Technical Support Center
TRM ..... Technical Requirements Manual

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

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

DBNPS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
HU1.1	HU1	1
HU1.2	HU1	2
HU1.3	HU1	3
HU2.1	HU2	1
HU3.1	HU3	1
HU3.2	HU3	2
HU3.3	HU3	3
HU3.4	HU3	4
HU4.1	HU4	1
HU4.2	HU4	2
HU4.3	HU4	3
HU4.4	HU4	4
HU7.1	HU7	1
HA1.1	HA1	1
HA1.2	HA1	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

DBNPS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
HG1.1	HG1	1
HG7.1	HG7	1
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
EU1.1	E-HU1	1
CU1.1	CU1	1
CU1.2	CU1	2

DBNPS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
CU2.1	CU2	1
CU3.1	CU3	1
CU3.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1
CU5.2	CU5	2
CU5.3	CU5	3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA3.1	CA3	1, 2
CA6.1	CA6	1
CS1.1	CS1	3
CG1.1	CG1	2
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	1
SU4.2	SU3	2
SU5.1	SU4	1
SU6.1	SU5	1
SU6.2	SU5	2
SU7.1	SU6	1
SU7.2	SU6	2
SU7.3	SU6	3
SU8.1	SU7	1

DBNPS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
SU8.2	SU7	2
SA1.1	SA1	1
SA3.1	SA2	1
SA6.1	SA5	1
SA9.1	SA9	1
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1
SG1.1	SG1	1
SG1.2	SG8	1
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1

7.0 ATTACHMENTS

Attachment 1, Emergency Action Level Technical Bases

Attachment 2, Fission Product Barrier Matrix and Basis

Attachment 3, Safe Shutdown Rooms/Areas Tables R-2 & H-2 Bases

Category H – Hazards and Other Conditions Affecting Plant Safety

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

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

1. Security

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

2. Seismic Event

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

3. Natural or Technology Hazard

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

4. Fire

FIREs can pose significant hazards to personnel and reactor safety. Appropriate for classification are FIREs within the site PROTECTED AREA or FIREs that may affect operability of equipment needed for safe shutdown.

5. Hazardous Gas

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

6. Control Room Evacuation

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

7. Emergency Director Judgment

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

category provide the Emergency Director the latitude to classify emergency conditions consistent with the established classification criteria based upon Emergency Director judgment.

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility

EAL:

HG1.1 General Emergency

A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor

AND EITHER of the following has occurred:

ANY of the following safety functions **cannot** be controlled or maintained

- Reactivity
- Core cooling (RCS inventory)
- RCS heat removal (ability to maintain heat sink)

OR

Damage to spent fuel has occurred or is IMMINENT

Mode Applicability:

ALL

NEI 99-01 Basis:

This IC addresses an event in which a HOSTILE FORCE has taken physical control of the facility to the extent that the plant staff can no longer operate equipment necessary to maintain key safety functions. It also addresses a HOSTILE ACTION leading to a loss of physical control that results in actual or IMMINENT damage to spent fuel due to 1) damage to a spent fuel pool cooling system (e.g., pumps, heat exchangers, controls, etc.) or, 2) loss of spent fuel pool integrity such that sufficient water level cannot be maintained.

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

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 DBNPS Physical Security Plan (ref.1).

DBNPS Basis:

None

DBNPS Basis Reference(s):

1. DBNPS Physical Security Plan (safeguards)
2. DB-OP-02544 Security Events or Threats (restricted)
3. RA-EP-02890 Emergency Response Organization Response to Security Events or Threats
4. NEI 99-01 HG1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA

EAL:

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

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

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

DBNPS Basis:

The Security Shift Supervision is defined as the Security Shift Supervisor or designee.

These individuals 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 is highly controlled due to the strict secrecy controls placed on the DBNPS Physical Security Plan (Safeguards) information (ref. 1).

DBNPS Basis Reference(s):

1. DBNPS Physical Security Plan (safeguards)
2. DB-OP-02544 Security Events or Threats (restricted)
3. RA-EP-02890 Emergency Response Organization Response to Security Events or Threats
4. NEI 99-01 HS1

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HA1.1 Alert

A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

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

~~EAL #2 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 procedure).~~

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

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

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

DBNPS Basis:

The Security Shift Supervision is defined as the Security Shift Supervisor or designee.

Escalation of the emergency classification level would be via IC HS1 or HG1.

DBNPS Basis Reference(s):

1. DBNPS Physical Security Plan (safeguards)
2. DB-OP-02544 Security Events or Threats (restricted)
3. RA-EP-02890 Emergency Response Organization Response to Security Events or Threats
4. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HA1.2	**Alert**
A validated notification from NRC of an aircraft attack threat within 30 min. of the site	

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

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

~~EAL #2~~This EAL addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-

related notifications are made in a timely manner so that plant personnel and OROs are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with (site-specific security procedures).

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

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

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

DBNPS Basis:

The Security Shift Supervision is defined as the Security Shift Supervisor or designee.

Escalation of the emergency classification level would be via IC HS1 or HG1.

DBNPS Basis Reference(s):

1. DBNPS Physical Security Plan (safeguards)
2. DB-OP-02544 Security Events or Threats (restricted)
3. RA-EP-02890 Emergency Response Organization Response to Security Events or Threats
4. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1 Unusual Event

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3, 4). Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

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

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

~~EAL #2 addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with (site-specific procedure)DBNPS.~~

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

Emergency plans and implementing procedures are public documents; therefore, EALs should

not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the DBNPS Physical Security Plan (ref. 1).

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

DBNPS Basis:

The Security Shift Supervision is defined as the Security Shift Supervisor or designee.

This EAL is based on the DBNPS Physical Security Plan (ref. 1).

DBNPS Basis Reference(s):

1. DBNPS Physical Security Plan (safeguards)
2. DB-OP-02544 Security Events or Threats (restricted)
3. RA-EP-02890 Emergency Response Organization Response to Security Events or Threats
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.2	**Unusual Event**
Notification of a credible security threat directed at the site	

Mode Applicability:

ALL

NEI 99-01 Basis:

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

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3, 4). Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

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

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

~~EAL #2~~This EAL addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with ~~(site-specific procedure)~~the DBNPS Physical Security Plan (ref. 1).

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

Emergency plans and implementing procedures are public documents; therefore, EALs should

not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the DBNPS Physical Security Plan (ref. 1).

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

DBNPS Basis:

The Security Shift Supervision is defined as the Security Shift Supervisor or designee.

This EAL is based on the DBNPS Physical Security Plan (ref. 1).

DBNPS Basis Reference(s):

1. DBNPS Physical Security Plan (safeguards)
2. DB-OP-02544 Security Events or Threats (restricted)
3. RA-EP-02890 Emergency Response Organization Response to Security Events or Threats
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.3	**Unusual Event**
A validated notification from the NRC providing information of an aircraft threat	

Mode Applicability:

ALL

NEI 99-01 Basis:

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

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3, 4). Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

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

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

~~EAL #2 addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with (site-specific procedure).~~

~~EAL #3~~This EAL addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with the DBNPS Physical Security Plan (ref. 1)~~(site-specific procedure)~~.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be

advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the DBNPS Physical Security Plan (ref. 1).

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

DBNPS Basis:

The Security Shift Supervision is defined as the Security Shift Supervisor or designee.

This EAL is based on the DBNPS Physical Security Plan (ref. 1).

DBNPS Basis Reference(s):

1. DBNPS Physical Security Plan (safeguards)
2. DB-OP-02544 Security Events or Threats (restricted)
3. RA-EP-02890 Emergency Response Organization Response to Security Events or Threats
4. NEI 99-01 HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE levels

EAL:

HU2.1	**Unusual Event**
Seismic event > OBE as indicated by OBE alarm on seismic alarm panel C5764A	

Mode Applicability:

ALL

NEI 99-01 Basis:

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

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event (e.g., lateral accelerations in excess of 0.08g). The Shift Manager or Emergency Director 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.

DBNPS Basis:

Ground motion acceleration of 0.08g horizontal or 0.053g vertical is the Maximum Probable Earthquake as is considered generically as the Operating Basis Earthquake for DBNPS (ref. 1).

Four triaxial strong motion accelerographs are installed within plant structures. A twelve-channel magnetic tape acceleration recording device is used to record all triaxial accelerometer transducer outputs. The recording device is located in the control room with a playback unit featuring a visual display. The time-history recorded at each accelerograph location can be analyzed to determine its corresponding peak acceleration values and to verify that site Operating Basis Earthquake (OBE) limits have not been exceeded. Immediate

control room alarm indication of an earthquake of 0.08 g or greater is annunciated on the seismic control panel (C5764A), following seismic trigger actuation by the Containment Concrete Foundation accelerograph (ref. 2).

RA-EP-02820 Earthquake provides the guidance for determining any required response actions if the OBE earthquake threshold is exceeded (ref. 3).

When the Seismic Monitoring Cabinet key locked switch is placed in OFF, ALL OBE and SSE alarms and cabinet recording functions (TNC 8.3.3 functions 1 and 3 and the DBRM-EMER-5003 Seismic Event Detection function) are nonfunctional. Therefore, the Compensatory Measures listed in DBRM-EMER-5003, Equipment Important to Emergency Response for a loss of the Seismic Event Detection function are required to be employed by the operators to determine if an earthquake occurs and to determine the magnitude of the event in a timely manner" (ref. 5).

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. However, such confirmation should not preclude a timely emergency declaration based on receipt of the OBE alarm. Contact the NEIC by calling **(303) 273-8500**. Select **option #1**, then **#2** and inform the analyst you wish to confirm recent seismic activity near DBNPS. Provide the analyst with the following DBNPS coordinates: **41º 35' 49" north latitude, 83º 05' 16" west longitude** and exact time of seismic activity (ref. 4). Alternatively, access near real-time seismic activity via the NEIC website:

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

DBNPS Basis Reference(s):

1. Updated FSAR Section 3.1 Seismic Design
2. Updated FSAR Section 3.7.4 Criteria for Seismic Instrumentation Program
3. RA-EP-02820 Earthquake
4. UFSAR Section 2.1.1 Site Location
5. CA 2012-04838-08
6. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

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

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

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

DBNPS Basis:

Response actions associated with a tornado onsite is provided in RA-EP-02810 Tornado or High Winds (ref. 1).

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

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

DBNPS Basis Reference(s):

1. RA-EP-02810 Tornado or High Winds
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.2 Unusual Event

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

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

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

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

DBNPS Basis:

Areas containing safe shutdown equipment susceptible to internal FLOODING are Service Water Pump Room and adjacent areas, Component Cooling Water Pump Room and Emergency Core Cooling System Room(s) (ref.1). Refer to EAL CA6.1 or SA9.1 for internal FLOODING affecting one or more SAFETY SYSTEM trains.

DBNPS Basis Reference(s):

1. RA-EP-02880 Internal Flooding
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 PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

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

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

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

DBNPS Basis:

As used here, the term "offsite" means the areas external to the DBNPS PROTECTED AREA.

DBNPS Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.4 Unusual Event

A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.~~EAL #5 addresses (site-specific description).~~

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

DBNPS Basis:

As used here, the term "onsite" means the areas within the DBNPS OWNER CONTROLLED AREA.

DBNPS Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – FIRE

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

EAL:

HU4.1 Unusual Event

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

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

AND

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

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

Table H-1 Safe Shutdown Fire Areas
• Containment • Control Room • Auxiliary Building • Intake Structure • Borated Water Storage Tank

Mode Applicability:

ALL

NEI 99-01 Basis:

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

~~EAL #1~~

~~The~~ For EAL HU4.1 the intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.~~EAL #2~~

~~This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.~~

~~A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.~~

~~If an actual FIRE is verified by a report from the field, then EAL #1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.~~

~~EAL #3~~

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

~~EAL #4~~

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

Basis-Related Requirements from Appendix R

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."
>
> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.
>
> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL ~~#2~~HU4.2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

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

DBNPS Basis:

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

Table H-1 Fire Areas are based on DBNPS Unit 1 Fire Hazard Analysis Report. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1, 2).

One of the following 4 things must occur within 15-minutes of the original FIRE indication / notification (ref. 3, 4):

1. The alarm must be proven to be spurious (there is no FIRE),
2. The FIRE must be extinguished,
3. The FIRE must be proven not to be in a Table H-1 area, or
4. The Unusual Event must be declared.

For the purposes of declaring an emergency event, the term "extinguished" means no visible flames.

The 15 minute time period starts with EITHER:

1. A credible notification to the Control Room that a FIRE is occurring in a table H-1 area, (verbal notification from the scene) or

2. Control Room indication of multiple fire detection system alarms/actuations in a Table H-1 area

DBNPS Basis Reference(s):

1. DBNPS Fire Hazard Analysis Report
2. DB-OP-02529 Fire Procedure
3. Condition Report 09-66994, Emergency Action Level Clarification
4. Condition Report 09-69475, White Finding Identified For Inadequate Emergency Classification of Event
5. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – FIRE

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

EAL:

HU4.2 Unusual Event

Receipt of a single fire alarm (i.e., no other indications of a FIRE)

AND

The fire alarm is indicating a FIRE within **ANY** Table H-1 area

AND

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

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

Table H-1 Safe Shutdown Fire Areas
• Containment • Control Room • Auxiliary Building • Intake Structure • Borated Water Storage Tank

Mode Applicability:

ALL

NEI 99-01 Basis:

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

~~EAL #1~~

~~The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.~~

~~Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.~~

~~EAL #2~~

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then ~~EAL #1~~HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

~~EAL #3~~

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

~~EAL #4~~

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

Basis-Related Requirements from Appendix R

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

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

DBNPS Basis:

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

Table H-1 Fire Areas are based on DBNPS-1465.00-00-0006 Design Basis Specification for the Plant Fire Protection and AP/0/A/5500/045 Plant Fire. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1, 2).

One of the following 4 things must occur within 30 minutes of the original FIRE indication / notification (ref. 3, 4):

1. The alarm must be proven to be spurious (there is no FIRE),
2. The FIRE must be extinguished,
3. The FIRE must be proven not to be in a Table H-1 area, or
4. The Unusual Event must be declared.

For the purposes of declaring an emergency event, the term "extinguished" means no visible

flames.

The 30 minute time period starts with Control Room indication of a fire detection system alarm/actuation (any single alarm or a single actuation of a system).

DBNPS Basis Reference(s):

1. DBNPS Fire Hazard Analysis Report
2. DB-OP-02529 Fire Procedure
3. Condition Report 09-66994, Emergency Action Level Clarification
4. Condition Report 09-69475, White Finding Identified For Inadequate Emergency Classification of Event
5. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – FIRE

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

EAL:

HU4.3 Unusual Event

A FIRE within the plant PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

~~EAL #1~~

~~The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.~~

~~Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.~~

~~EAL #2~~

~~This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.~~

~~A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to~~

~~determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.~~

~~If an actual FIRE is verified by a report from the field, then EAL #1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.~~

~~EAL #3~~

In addition to a FIRE addressed by EAL ~~#1~~HU4.1 or ~~EAL #2~~HU4.2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety. ~~This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA. [*Sentence for plants with an ISFSI outside the plant Protected Area*]EAL #4~~

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

~~Basis-Related Requirements from Appendix R~~

~~Appendix R to 10 CFR 50, states in part:~~

> ~~Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."~~
>
> ~~When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.~~
>
> ~~Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.~~

~~In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.~~

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

DBNPS Basis:

None

DBNPS 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 plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

Mode Applicability:

ALL

NEI 99-01 Basis:

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

~~EAL #1~~

~~The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.~~

~~Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.~~

~~EAL #2~~

~~This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.~~

~~A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.~~

~~If an actual FIRE is verified by a report from the field, then EAL #1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.~~

~~EAL #3~~

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

~~EAL #4~~

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

~~Basis-Related Requirements from Appendix R~~

~~Appendix R to 10 CFR 50, states in part:~~

> ~~Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."~~
>
> ~~When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.~~
>
> ~~Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.~~

~~In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.~~

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

DBNPS Basis:

None

DBNPS 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 Safe Shutdown 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 Shutdown Rooms/Areas

Room/Area	Mode Applicability
Aux Bldg. 565' ele. Room 236 #2 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 585' ele. Room 304 corridor outside #3 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 603' ele. Room 427 - #2 Electrical Penetration Room	1, 2, 3

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 – Hot Standby

NEI 99-01 Basis:

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

An Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.

Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the Emergency Director judgment that the gas concentration in the affected room/area is sufficient

to preclude or significantly IMPEDE procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as IMPEDED if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

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

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures, which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or IMPEDE a required action.
- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death. A steam leak of significant magnitude that prevents access to the area should be considered an asphyxiant.

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.~~.

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

DBNPS Basis:

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

The list of plant rooms or areas with entry-related mode applicability identified specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or

areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

DBNPS Basis Reference(s):

1. Attachment 3 Safe Shutdown Rooms/Areas Tables R-2 & H-2 Bases
2. NEI 99-01 HA5

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

Initiating Condition: Inability to control a key safety function from outside the Control Room

EAL:

HS6.1 Site Area Emergency

An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel

AND

Control of **ANY** of the following key safety functions is not reestablished within 15 min. (Note 1):

- Reactivity
- Core Cooling (RCS inventory)
- RCS heat removal (ability to maintain heat sink)

Note 1: The Emergency Director 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

NEI 99-01 Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not “control” is established at the remote safe shutdown location(s) is based on Emergency Director judgment. The Emergency Director is expected to make a reasonable, informed judgment within ~~(the site-specific time for transfer)~~15-minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

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

DBNPS Basis:

Shift Manager determines the Control Room requires evacuation. Control Room may become uninhabitable by FIRE, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (ref. 1, 2).

The 15-minute time for transfer is based on analysis or assessments as to how quickly control must be reestablished without core uncovering and/or core damage. The 15-minute time period starts when EITHER 1) control of the plant is no longer maintained in the Control Room or 2) the last operator has left the Control Room, whichever comes first.

DBNPS Basis Reference(s):

1. DB-OP-02508 Control Room Evacuation
2. DB-OP 02519 Serious Control Room Fire
3. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

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

EAL:

HA6.1 Alert

An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel

Mode Applicability:

ALL

NEI 99-01 Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

This EAL is only applicable when the decision has been made to evacuate the Control Room, and not when conditions that may require evacuation are being evaluated by referring to either DB-OP-02519 or DB-OP-02508 (ref. 1, 2).

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

DBNPS Basis:

Shift Manager (SM) determines Control Room requires evacuation. Control Room inhabitability may be caused by FIRE, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (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.

DBNPS Basis Reference(s):

1. DB-OP-02508 Control Room Evacuation
2. DB-OP 02519 Serious Control Room Fire
3. NEI 99-01 HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency

EAL:

HG7.1	**General Emergency**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	

Mode Applicability:

ALL

NEI 99-01 Basis:

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

DBNPS Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the DBNPS Emergency Response Plan. The Operations Shift Manager(SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, 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 (ref. 1).

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

DBNPS Basis Reference(s):

1. DBNPS Emergency Plan section 5.2 DBNPS Emergency Management
2. NEI 99-01 HG7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency

EAL:

HS7.1 Site Area Emergency

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. **ANY** releases are **not** expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the SITE BOUNDARY.

Mode Applicability:

ALL

NEI 99-01 Basis:

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

DBNPS Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the DBNPS Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, 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 (ref. 1).

DBNPS Basis Reference(s):

1. DBNPS Emergency Plan section 5.2 DBNPS Emergency Management
2. NEI 99-01 HS7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

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

EAL:

HA7.1 Alert

Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. **ANY** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Mode Applicability:

ALL

NEI 99-01 Basis:

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

DBNPS Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the DBNPS Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, 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 (ref.1).

DBNPS Basis Reference(s):

1. DBNPS Emergency Plan section 5.2 DBNPS Emergency Management
2. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a UE

EAL:

HU7.1 Unusual Event

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Mode Applicability:

ALL

NEI 99-01 Basis:

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

DBNPS Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the DBNPS Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, 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 (ref. 1).

DBNPS Basis Reference(s):

1. DBNPS Emergency Plan section 5.2 DBNPS Emergency Management
2. NEI 99-01 HU7

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 radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

EAL:

RG1.1	**General Emergency**
Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E +00 µCi/cc for 15 minutes or longer (Notes 1, 2, 3, 4)	

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

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

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

DBNPS Basis:

The DBNPS gaseous effluent limits for RG1.1 are based on values that equate to an offsite dose greater than EITHER:

- 1000 mrem TEDE
- 5000 mrem CDE Thyroid

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

The station vent is the primary effluent monitor release point. Other pathways such as Safety and relief valves, as well as AFW and MSSV, do not have direct effluent release monitors. Releases from these points would be verified and monitored by RG1.3 survey readings.

The “GE” Station Vent gaseous effluent monitor value corresponds to calculated doses of 100% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) per EP-EALCALC-DB-0703 Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1 and RU1) (ref. 1).

DBNPS Basis Reference(s):

1. EP-EALCALC-DB-0703 Rev. 2 Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1 and RU1)
2. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

DBNPS Basis:

Dose assessments are performed by computer-based methods (ref. 1, 2).

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

DBNPS Basis Reference(s):

1. NOP-LP-5402 Davis-Besse MIDAS Dose Assessment Software
2. RA-EP-02240 Offsite Dose Assessment
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.3 General Emergency

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

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

(Notes 1, 2)

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

DBNPS Basis:

NOP-LP-5015 FENOC Field Monitoring Teams Radiation Monitoring Teams Field Surveys provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

DBNPS Basis Reference(s):

1. NOP-LP-5015 FENOC Field Monitoring Teams Radiation Monitoring Teams Field Surveys
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 100 mrem TEDE or 500 mrem thyroid CDE

EAL:

RS1.1 Site Area Emergency

Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E -01 µCi/cc for 15 minutes or longer (Notes 1, 2, 3, 4)

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

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

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

DBNPS Basis:

The DBNPS gaseous effluent limits for RS1.1 are based on values that equate to an offsite dose greater than EITHER:

- 100 mrem TEDE
- 500 mrem CDE Thyroid

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

The station vent is the primary effluent monitor release point. Other pathways such as Safety and relief valves, as well as AFW and MSSV, do not have direct effluent release monitors. Releases from these points would be verified and monitored by RS1.3 survey readings.

The “SAE” Station Vent gaseous effluent monitor value corresponds to calculated doses of 10% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) per EP-EALCALC-DB-0703 Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1 and RU1) (ref. 1).

DBNPS Basis Reference(s):

1. EP-EALCALC-DB-0703 Rev.2 Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1 and RU1)
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

DBNPS Basis:

Dose assessments are performed by computer-based or manual methods (ref. 1, 2).

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

DBNPS Basis Reference(s):

1. NOP-LP-5402 Davis-Besse MIDAS Dose Assessment Software
2. RA-EP-02240 Offsite Dose Assessment
3. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.3 Site Area Emergency

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

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

(Notes 1, 2)

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

~~longer valid for classification purposes.~~

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

DBNPS Basis:

NOP-LP-5015 FENOC Field Monitoring Teams Radiation Monitoring Teams Field Surveys provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

DBNPS Basis Reference(s):

1. NOP-LP-5015 FENOC Field Monitoring Teams Radiation Monitoring Teams Field Surveys
2. NEI 99-01 AS1

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

Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E -02 µCi/cc for 15 minutes or longer (Notes 1, 2, 3, 4)

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

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

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

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

DBNPS Basis:

The DBNPS gaseous effluent limits for RA1.1 are based on values that equate to an offsite dose greater than EITHER:

- 10 mrem TEDE
- 50 mrem CDE Thyroid

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

The “Alert” Station Vent gaseous effluent monitor value corresponds to calculated doses of 1% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) per EP-EALCALC-DB-0703 Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1 and RU1) (ref. 1).

The station vent is the primary effluent monitor release point. Other pathways such as Safety and relief valves, as well as AFW and MSSV, do not have direct effluent release monitors. Releases from these points would be verified and monitored by RA1.3 survey readings.

DBNPS Basis Reference(s):

1. EP-EALCALC-DB-0703 Rev.2 Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1 and RU1)
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.2 Alert

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

DBNPS Basis:

Dose assessments are performed by computer-based and manual methods (ref. 1, 2).

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

DBNPS Basis Reference(s):

1. NOP-LP-5402 Davis-Besse MIDAS Dose Assessment Software
2. RA-EP-02240 Offsite Dose Assessment
3. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.3 Alert

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

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

DBNPS Basis:

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

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

DBNPS Basis Reference(s):

1. DBNPS ODCM
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.4 Alert

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

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

(Notes 1, 2)

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

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

DBNPS Basis:

NOP-LP-5015 FENOC Field Monitoring Teams Radiation Monitoring Teams Field Surveys provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

Note: EAL thresholds reflect the State of Ohio guidance to utilities within their jurisdiction to evaluate the consequences of radiological releases in terms of a Child Thyroid PAG rather than an EPA-400 CDE Thyroid PAG.

DBNPS Basis Reference(s):

1. NOP-LP-5015 FENOC Field Monitoring Teams Radiation Monitoring Teams Field Surveys
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.1 Unusual Event

Station Vent Channel 1 Noble Gas (RE 4598AA/BA) > 5.72E-03 µCi/cc for 60 minutes or longer

OR

ANY of the following effluent monitors > 2 times the high alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer:

- Waste Gas System Outlet (RE 1822A or B).
- Clean Waste System Outlet (RE 1770A or B).
- Miscellaneous Waste System Outlet (RE 1878A or B).
- Discharge permit specified monitor.

(Notes 1, 2, 3)

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

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

This IC addresses a potential decrease in the level of safety of the plant as indicated by a low-level radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release). It includes ANY gaseous or liquid radiological release,

monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.

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

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

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

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

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

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

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

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

DBNPS Basis:

The “UE” gaseous and liquid release values represent two times the appropriate ODCM release rate limits associated with the specified monitors (ref. 2).

Gaseous Releases

Instrumentation that may be used to assess this EAL is listed below (ref. 1):

- Station Vent – RE 4598AA or BA
- Waste Gas System Outlet – RE 1822 A or B (batch release)

The Station Vent gaseous effluent (RE 4598AA/BA) value of 2 times the ODCM setpoint was determined using formulas, isotopic dose conversion factors and meteorology data as specified in the ODCM based on a normal operating isotopic mixture (no clad damage condition). Assumptions and calculation inputs are provided in EP-EALCALC-DB-0703 Rev. 2 (ref. 2).

The Waste Gas Outlet (RE 1822 A or B for batch release) are the release monitors normally used for planned discharges. If a discharge is performed using a different flow path or effluent monitor (e.g., a portable or temporary effluent monitor), then the declaration criteria will be based on the monitor specified in the discharge permit.

Liquid Releases

Instrumentation that may be used to assess this EAL is listed below (ref. 1):

- Clean Waste System Outlet – RE 1770 A or B (batch release)
- Misc. Waste System Outlet – RE 1878 A or B (batch release)

The liquid release monitors listed are those normally used for planned discharges. If a discharge is performed using a different flow path or effluent monitor (e.g., a portable or temporary effluent monitor), then the declaration criteria will be based on the monitor specified in the discharge permit.

DBNPS Basis Reference(s):

1. DBNPS ODCM
2. EP-EALCALC-DB-0703 Rev. 2 Radiological Gaseous Effluent EAL Values (EALs RG1, RS1, RA1 and RU1)
3. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.2 Unusual Event

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

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

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

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

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

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

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

DBNPS Basis:

Grab samples are used to determine release concentrations or release rates, confirm meter readings, or indicate the need for sampling when the effluent monitors are not in service or other alarms occur. The maximum instantaneous release rate limits are calculated in accordance with the ODCM. These are indicated on approved discharge permit release packages (ref. 1).

DBNPS Basis Reference(s):

1. DBNPS ODCM
2. NEI 99-01 AU1

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. for ≥ 60 min. (Note 1)	

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

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

DBNPS Basis:

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

Indicated level of 1 ft. on LI4801A (primary) or LI4801B (Backup) corresponds approximately to 1 ft. above the top of the SFP racks (ref. 2).

DBNPS Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. Engineering Change Package 13-0596
3. NEI 99-01 AG2

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.	

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

DBNPS Basis:

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

Indicated level of 1 ft. on LI4801A (primary) or LI4801B (Backup) corresponds approximately to 1 ft. above the top of the SFP racks (ref. 2).

DBNPS Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. Engineering Change Package 13-0596
3. NEI 99-01 AS2

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

NEI 99-01 Basis:

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

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

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

EAL #1

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

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

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

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

~~EAL #3~~

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

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

DBNPS Basis:

None

DBNPS Basis Reference(s):

1. DB-OP-02003 3-1-B, SFP LVL
2. DB-OP-02003 3-1-A, REFUELING CANAL LVL
3. DB-OP-02547 Spent Fuel Pool Cooling Malfunctions
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:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity as indicated by a high radiation alarm on **ANY** of the following radiation monitor indications:

- RE 8426 SFP Area
- RE 8427 SFP Area
- RE 8417 Fuel Handling Area
- RE 8418 Fuel Handling Area
- RE 8425 Equipment Hatch Area
- RE 8446/8447 Fuel Handling Exhaust
- RE 4598AA/BA Station Vent

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

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

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

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

DBNPS Basis:

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

The high alarm setpoints for the radiation monitors are set to be indicative of significant increases in area and/or airborne radiation (ref. 2).

DBNPS Basis Reference(s):

1. DB-OP-02530 Fuel Handling Accident
2. Radiation Setpoints Manual
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:

RA2.3 **Alert**
Lowering of spent fuel pool level to 10 ft.

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

~~This EAL addresses a release of radioactive material caused by mechanical damage to irradiated fuel. Damaging events may include the dropping, bumping or binding of an assembly, or dropping a heavy load onto an assembly. A rise in readings on radiation~~

~~monitors should be considered in conjunction with in-plant reports or observations of a potential fuel damaging event (e.g., a fuel handling accident).~~

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

Escalation of the emergency classification level would be via IC~~s~~ ~~AS1~~ RS1 or ~~AS2 (*see AS2 Developer Notes*)~~RS2.

DBNPS Basis:

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

Indicated level of 10 ft. on LI4801A (primary) or LI4801B (Backup) corresponds to plant elevation 588 ft. 6 in. (ref. 2).

DBNPS Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. Engineering Change Package 13-0596
3. NEI 99-01 AA2

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 **ANY** of the following:

- SFP level alarm or indication (LI 1600)
- SFP level indication (LI 4801 A/B)
- Refueling Canal low water level alarm or indication (LI 1627)
- Visual observation

AND

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

- RE 8426 SFP Area
- RE 8427 SFP Area
- RE 8417 Fuel Handling Area
- RE 8418 Fuel Handling Area
- RE 8425 Equipment Hatch Area

Mode Applicability:

ALL

NEI 99-01 Basis:

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

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

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

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

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

DBNPS Basis:

The SFP low water level alarm setpoint is actuated by LS 1600 at a setpoint of 23.2 feet on LI 1600 (ref. 1).

The refueling canal low water level alarm setpoint is actuated by LS 1627 at a setpoint of 23.4 feet on LI 1627 (this corresponds to 1.5 inches below normal) (ref. 2).

The LI 4801 A/B is indication only, with no alarm or computer point function.

Water level restoration instructions are performed in accordance with AOPs (ref. 1, 2, 3).

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

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

DBNPS Basis Reference(s):

1. DB-OP-02003 3-1-B, SFP LVL
2. DB-OP-02003 3-1-A, REFUELING CANAL LVL
3. DB-OP-02547 Spent Fuel Pool Cooling Malfunctions
4. NEI 99-01 AU2

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 (RE 8430 or 8431)
- Central Alarm Station (RE 8435 or 8436)

Mode Applicability:

ALL

NEI 99-01 Basis:

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

~~An emergency declaration is not warranted if any of the following conditions apply.~~

- ~~The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the elevated radiation levels). For example, the plant is in Mode 1 when the radiation increase occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.~~

- ~~The increased radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).~~

- ~~The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).~~
- ~~The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.~~

Escalation of the emergency classification level would be via Recognition Category ~~A~~R, C or F ICs.

DBNPS Basis:

Areas that meet this threshold include the Control Room and the Central Alarm Station (CAS). RE 8430 and 8431 monitors the Control room for area radiation. The CAS area does not have installed radiation monitoring but RE 8435 and 8436 are in near proximity and can be used to approximate CAS area radiation levels (ref. 1, 2). The CAS is included because of its importance to permitting access to areas required to assure safe plant operations.

DBNPS Basis Reference(s):

1. DB-OP-06412 Process and Area Radiation Monitor
2. SD-017B System Description for Area Radiation Monitors
3. 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 Safe Shutdown 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 Shutdown Rooms/Areas

Room/Area	Mode Applicability
Aux Bldg. 565' ele. Room 236 #2 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 585' ele. Room 304 corridor outside #3 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 603' ele. Room 427 - #2 Electrical Penetration Room	1, 2, 3

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 – Hot Standby

NEI 99-01 Basis:

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or IMPEDE personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Emergency Director should consider the cause of the increased radiation levels and determine if another IC may be applicable.

For ~~EAL #2~~RA3.2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the increased radiation levels. Access should be considered as IMPEDED if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., installing temporary shielding, requiring use of non-routine protective equipment, requesting an extension in dose limits beyond normal administrative limits).

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.

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.

Escalation of the emergency classification level would be via Recognition Category ~~A~~R, C or F ICs.

DBNPS Basis:

If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

The list of plant rooms or areas with entry-related mode applicability identified specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

DBNPS Basis Reference(s):

1. Attachment 3 Safe Shutdown Rooms/Areas Tables R-2 & H-2 Bases
2. RA-EP-02861 Radiological Incidents
3. NEI 99-01 AA3

Category E – Dry Fuel Storage Facility (DFSF)

EAL Group: ANY (EALs in this category are applicable to ANY plant condition, hot or cold)

A dry fuel storage facility (DFSF) is a complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage. A significant amount of the radioactive material contained within a cask/canister must escape its packaging and enter the biosphere for there to be a significant environmental effect resulting from an accident involving the dry storage of spent nuclear fuel.

An Unusual Event is declared based on the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated.

Minor surface damage that does not affect storage cask/canister boundary is excluded from the scope of these EALs.

Category: E - DFSF

Sub-category: None

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Notification of Unusual Event

Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel HSM cask > **EITHER** of the following:

- 100 mrem/hr (neutron + gamma) on the HSM cask wall or roof
- 100 mrem/hr (neutron + gamma) on the center of the HSM cask door

Mode Applicability:

ALL

NEI 99-01 Basis:

This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes, which could cause challenges in removing the cask or fuel from storage.

The existence of “damage” is determined by radiological survey. The technical specification multiple of “2 times”, which is also used in Recognition Category ~~A~~ R IC R~~A~~U1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the “on-contact” dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

Security-related events for DFSF~~ISFSI~~s are covered under ICs HU1 and HA1.

DBNPS Basis:

The DBNPS DFSF utilizes one design for dry spent fuel storage: the Nuclear Horizontal Modular Storage (NUHOMS) system.

The system consists of Horizontal Storage Modules (HSM) that contains dry shielded canisters (DSCs). The DSC serves as the CONFINEMENT BOUNDARY. The DSC is welded and designed to provide confinement of all radionuclides under normal, off normal, and accident conditions.

CONFINEMENT BOUNDARY is defined as the barrier(s) between areas containing radioactive substances and the environment. Therefore, damage to a CONFINEMENT BOUNDARY must be a confirmed physical breach between the spent fuel and the environment for the DSC.

The values shown represent 2 times the limits specified in the DFSF Site CSAR for radiation external to a loaded DSC (ref. 1).

DBNPS Basis Reference(s):

1. Site CSAR NUH-003 Table 7.3-2 Shielding Analysis Results for 5 year Cooled Fuel NUHOMS-24P System
3. NEI 99-01 E-HU1

Category C – Cold Shutdown / Refueling System Malfunction

EAL Group: Cold Conditions (RCS temperature ≤ 200ºF); EALs in this category are applicable only in one or more cold operating modes.

Category C EALs are directly associated with Cold Shutdown or Refueling system safety functions. Given the variability of plant configurations (e.g., systems out-of-service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The Cold Shutdown and Refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, CONTAINMENT CLOSURE, and fuel clad integrity for the applicable operating modes (5 - Cold Shutdown, 6 - Refueling, 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 Essential AC Power

Loss of essential 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 4160 VAC essential buses.

3. RCS Temperature

Uncontrolled or inadvertent temperature or pressure increases are indicative of a potential loss of safety functions.

4. Loss of Essential 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 125 VDC buses.

5. Loss of Communications

Certain events that degrade plant operator's ability to communicate with essential personnel within or external to the plant warrant emergency classification.

6. Hazardous Event Affecting SAFETY SYSTEMS

Certain hazardous natural and technological events may result in visible damage to or degraded performance of SAFETY SYSTEMS warranting classification.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory affecting fuel clad integrity with containment challenged

EAL:

CG1.1 General Emergency

RCS level **cannot** be monitored for ≥ 30 min. (Note 1)

AND

Core uncovery is indicated by **ANY** of the following:

- UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT levels of sufficient magnitude to indicate core uncovery
- Containment Radiation Monitor (RE 4596A or B) reading > 16 R/hr
- Refueling Bridge Portable Area Radiation Monitor reading > 30 R/hr
- Erratic Source Range Monitor indication

AND

ANY Containment Challenge indication, Table C-1

Note 1: The Emergency Director 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 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

NEI 99-01 Basis:

This IC addresses the inability to restore and maintain ~~reactor vessel~~RCS level above the top of active fuel with containment challenged. This condition represents actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in ~~reactor vessel~~RCS level. If 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 [*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/R~~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*.

DBNPS Basis:

The lowest measurable RCS level is the elevation of the RCS hot leg mid-loop (0.4 ft. on LI 10596). Therefore, RCS inventory loss relative to the RCS level elevation corresponding to the top of active fuel must be detected by indirect leakage indications (ref. 1, 2, 3). Sump level increases must be evaluated against other potential sources of 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 in areas outside the containment that cannot be isolated, or a sump sample verified by Chemistry, could also be indicative of a loss of RCS inventory (ref. 1).

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed and portable area radiation monitors. RE 4596A or B, Containment Radiation Monitors are located in the containment, a reading greater than 16 R/hr is indicative of core uncovery. The Refuel Bridge Portable Radiation Monitor, when installed during refueling operations, is designed to provide monitoring of radiation due to a fuel handling event or loss of shielding during refueling operations. If this radiation monitor reaches and exceeds 30 R/hr, a loss of inventory indicative of core uncovery has occurred (ref. 2, 3).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.

Three conditions are associated with a challenge to containment integrity:

- CONTAINMENT CLOSURE is not established (Ref. 4).
- In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gases in the containment. However, containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. An explosive mixture can be formed when hydrogen gas concentration in the containment atmosphere is greater than 4% by volume in the presence of oxygen. Hydrogen monitors, although available at all times, are not in service during normal operations. They are started per DB-OP-02000 Table 3. Action is required if the measured concentration reaches 0.6% (notify TSC) and 3% (notify TSC) (ref 5).
- ANY UNPLANNED increase in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of CONTAINMENT CLOSURE capability. UNPLANNED containment pressure increases indicates CONTAINMENT CLOSURE cannot be assured and the containment cannot be relied upon as a barrier to fission product release.

DBNPS Basis Reference(s):

1. DB-OP-02527 Loss of Decay Heat Removal
2. EP-EALCALC-DB-0704 Radiation Monitor Readings for Core Uncovery During Refueling
3. DB-HP-01152 Performance of High Exposure Work
4. DB-OP-06904, Shutdown Operations
5. DB-OP-02000 Table 3, Containment Monitoring and Control
6. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory affecting core decay heat removal capability

EAL:

CS1.1 Site Area Emergency

RCS level cannot be monitored for ≥ 30 min. (Note 1)
AND
Core uncovery is indicated by **ANY** of the following:

- UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT levels of sufficient magnitude to indicate core uncovery
- Containment Radiation Monitor (RE 4596A or B) reading > 16 R/hr
- Refueling Bridge Portable Area Radiation Monitor reading > 30 R/hr
- Erratic Source Range Monitor indication

Note 1: The Emergency Director 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

NEI 99-01 Basis:

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~~RCS level. If ~~RCS/reactor vessel~~RCS level cannot be restored, fuel damage is probable.

~~Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS/reactor vessel levels of EALs 1.b and 2.b reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.~~

~~In EAL 3.a, t~~The 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor ~~(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.

Escalation of the emergency classification level would be via IC CG1 or ~~AG1~~RG1.

DBNPS Basis:

The lowest measurable RCS level is the elevation of the RCS hot leg mid-loop (0.4 ft. on LI 10596). Therefore, RCS inventory loss relative to the RCS level elevation corresponding to the top of active fuel must be detected by indirect leakage indications (ref. 1, 2, 3). Sump level increases must be evaluated against other potential sources of 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 in areas outside the containment that cannot be isolated, or a sump sample verified by Chemistry, could also be indicative of a loss of RCS inventory (ref. 1).

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed and portable area radiation monitors. RE 4596A or B, Containment Radiation Monitors are located in the containment, a reading greater than 16 R/hr is indicative of core uncovery. The Refuel Bridge Portable Radiation Monitor, when installed during refueling operations, is designed to provide monitoring of radiation due to a fuel handling event or loss of shielding during refueling operations. If this radiation monitor reaches and exceeds 30 R/hr, a loss of inventory indicative of core uncovery has occurred (ref. 2, 3).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.

DBNPS Basis Reference(s):

ATTACHMENT 1
EAL Bases

1. DB-OP-02527 Loss of Decay Heat Removal
2. EP-EALCALC-DB-0704 Radiation Monitor Readings for Core Uncovery During Refueling
3. DB-HP-01152 Performance of High Exposure Work
4. NEI 99-01 CS1

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 RCS level ≤ 0.4 ft. (LI 10596)	

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

NEI 99-01 Basis:

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

For this EAL ~~#1~~, a lowering of RCS water level at or below 0.4 ft.~~(site-specific level) ft.~~ 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, 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~~ RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

~~For EAL #2, the inability to monitor (reactor vessel/RCS [PWR] or RPV [BWR]) level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the (reactor vessel/RCS [PWR] or RPV [BWR]).~~

~~The 15-minute duration for the loss of level indication was chosen because it is half of the EAL duration specified in IC CS1~~

If RCS ~~the (reactor vessel/RCS [PWR] or RPV [BWR]) inventory~~ water level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

DBNPS Basis:

RCS level cannot be measured below the 571 feet elevation (0.4 feet on LI 10596) which is centerline of the hot leg inlet. Should RCS level drop below this point it is assumed water level cannot be monitored other than visually. Continued loss of RCS inventory may result in DHR pump cavitation (ref. 1).

DBNPS Basis Reference(s):

1. DB-OP-02527 Loss of Decay Heat Removal
2. 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 level cannot be monitored for ≥ 15 min. (Note 1)

AND EITHER

- UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

Note 1: The Emergency Director 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

NEI 99-01 Basis:

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

~~For EAL #1, a lowering of water level below (site-specific level) indicates that operator actions have not been successful in restoring and maintaining (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.~~

~~Although related, EAL #1 is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Residual Heat Removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.~~

For this EAL ~~#2~~, the inability to monitor ~~(reactor vessel/~~RCS ~~[PWR] or RPV [BWR])~~ level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the ~~(reactor vessel/RCS~~RCS ~~[PWR] or RPV [BWR])~~.

The 15-minute duration for the loss of level indication was chosen because it is half of the EAL duration specified in IC CS1.

If the ~~(reactor vessel/~~RCS~~ [PWR] or RPV [BWR])~~ inventory level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

DBNPS Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available. In the Refuel mode, the RCS is not intact and RCS level may be monitored by different means, including the ability to monitor level visually.

RCS level in the Refueling mode is normally monitored using the following instruments:

- LI 10596
- LI10577 A and B
- Clear tubing used for manometer level indication at the RCS cold legs.

RCS level indications (LI 10596, LI 10577A and B and cold leg tubing) provide accurate indication of water level when the RCS is at atmospheric pressure. The reactor vessel flange is at 577 ft. 7 in. (80 in. indicated) (ref. 1).

In this EAL, all RCS water level indication would be unavailable for greater than 15-minutes, and the RCS inventory loss must be detected by indirect leakage indications (ref. 1). Sump and tank level increases must be evaluated against other potential sources of 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, or a sump sample verified by Chemistry, could also be indicative of a loss of RCS inventory.

DBNPS Basis Reference(s):

1. DB-OP-06904 Shutdown Operations
2. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory for 15 minutes or longer

EAL:

CU1.1	**Unusual Event**
UNPLANNED loss of reactor coolant results in RCS level less than a required lower limit for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director 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

NEI 99-01 Basis:

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor ~~(reactor vessel/~~RCS~~[*PWR*] or RPV [*BWR*])~~ level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

~~Refueling~~ RCS draining evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

This EAL ~~#1~~ recognizes that the minimum required RCS ~~(reactor vessel/RCS [*PWR*] or RPV [*BWR*])~~ level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.

The 15-minute threshold duration allows sufficient time for prompt operator actions to restore and maintain the expected water level. This criterion excludes transient conditions causing a brief lowering of water level.

~~EAL #2 addresses a condition where all means to determine (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the (reactor vessel/RCS [*PWR*] or RPV [*BWR*]).~~

Continued loss of RCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA3.

DBNPS Basis:

With the plant in Cold Shutdown, RCS water level is normally maintained above a lower level band limit (ref. 1, 2), typically above the pressurizer low level setpoint. However, if RCS level is being controlled below the pressurizer low level setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

With the plant in Refueling mode, RCS water level is normally maintained at or above the reactor vessel flange, (Technical Specification LCO 3.9.6 requires at least 23 ft. of water above the top of the reactor vessel flange in the refueling canal when irradiated fuel is moved in containment). The reactor vessel flange is at 577 ft. 7 in. (80 in. indicated) (ref. 3, 4).

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available. RCS level in the Refueling mode is normally monitored using the following instruments:

- LI 10596
- LI 10577 A and B
- Clear tubing used for manometer level indication at the RCS cold legs.

RCS level indications (LI 10596, LI 10577A and B and cold leg tubing) provide accurate indication of water level when the RCS is at atmospheric pressure (ref. 3).

DBNPS Basis Reference(s):

1. DB-OP-02513 Pressurizer System Abnormal Operation
2. DB-OP-02004 4-2-E PZR LVL LO
3. DB-OP-06904 Shutdown Operations
4. DBNPS Technical Specifications Section 3.9.6 Refueling Canal Water Level
5. DB-OP-02527 Loss of Decay Heat Removal
6. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory for 15 minutes or longer

EAL:

CU1.2 Unusual Event

RCS level **cannot** be monitored

AND EITHER

- UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

NEI 99-01 Basis:

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor ~~(reactor vessel/~~RCS ~~[PWR] or RPV [BWR])~~ level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

~~Refueling~~ RCS draining evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

~~EAL #1 recognizes that the minimum required (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.~~

~~The 15-minute threshold duration allows sufficient time for prompt operator actions to restore and maintain the expected water level. This criterion excludes transient conditions causing a brief lowering of water level.~~

This EAL ~~#2~~ addresses a condition where all means to determine ~~(reactor vessel/~~RCS ~~[PWR] or RPV [BWR])~~ level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. 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/R~~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.

DBNPS Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available. RCS level in the Refueling mode is normally monitored using the following instruments:

- LI 10596
- LI10577 A and B
- Clear tubing used for manometer level indication at the RCS cold legs.

RCS level indications (LI 10596, LI 10577A and B and cold leg tubing) provide accurate indication of water level when the RCS is at atmospheric pressure. The reactor vessel flange is at 577 ft. 7 in. (80 in. indicated) (ref. 1).

In this EAL, all water level indication is unavailable and the RCS inventory loss must be detected by indirect leakage indications (ref. 1). Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. 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, or a sump sample verified by Chemistry, could also be indicative of a loss of RCS inventory.

DBNPS Basis Reference(s):

1. DB-OP-06904 Shutdown Operations
2. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Essential AC Power

Initiating Condition: Loss of **ALL** offsite and **ALL** onsite AC power to essential buses for 15 minutes or longer

EAL:

CA2.1 Alert

Loss of **ALL** offsite and **ALL** onsite AC power capability, Table C-2, to essential 4160V buses C1 and D1 for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-2 Offsite/Onsite AC Power Sources
Offsite: • X01 • X02 • X11 (back-fed via Main Transformer) **Onsite:** • EDG1 • EDG2 • SBODG

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, D - Defueled

NEI 99-01 Basis:

This IC addresses a total loss of AC power that compromises the performance of ALL SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink.

When in the Cold Shutdown, Refueling, or Defueled mode, this condition is not classified as a Site Area Emergency because of the increased time available to restore an essential~~emergency~~ bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in

these modes, this condition represents an actual or potential substantial degradation of the level of safety of the plant.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via IC CS1 or ~~AS1~~RS1.

DBNPS Basis:

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses C1 and D1 (ref. 1).

The essential buses during plant operation are normally powered from the 13.8KV offsite power system through their respective 13.8KV/4160V bus tie transformers, via the Unit Auxiliary Transformer (X11). In cold operating modes, the essential buses may be back-fed via the X11 and Main Transformer provided the main generator lead disconnect links are removed (ref. 1, 2). Credit for the X11 back-feed can only be taken if already aligned, as it takes greater than 15 minutes to align.

A standby source of offsite power to each 4160V essential bus is provided from the 13.8KV offsite power system via two separate and independent 13.8KV/4160V Startup Transformers (X01 & X02). Normally each startup transformer serves as a reserve power source for only one essential bus. However, a single startup transformer can be aligned to power both essential buses (ref. 1, 2).

Each essential bus has a diesel generator (EDG1 & EDG2) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power (ref. 1).

An Alternate AC power source, the SBO Diesel Generator, is located onsite. This onsite AC power source can be started from the Control Room and be loaded within 10 minutes of a SBO (ref. 1).

This cold condition EAL is equivalent to the hot condition loss of ALL offsite AC power EAL SS1.1.

DBNPS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. DB-OP-02521 Loss of AC Bus Power Sources
3. NEI 99-01 CA2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Essential AC Power

Initiating Condition: Loss of **ALL** but one AC power source to essential buses for 15 minutes or longer

EAL:

CU2.1 Unusual Event

AC power capability, Table C-2, to essential 4160V buses C1 and D1 reduced to a single power source for ≥ 15 min. (Note 1)

AND

ANY additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS

Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-2 Offsite/Onsite AC Power Sources
Offsite: • X01 • X02 • X11 (back-fed via Main Transformer) **Onsite:** • EDG1 • EDG2 • SBODG

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling, D - Defueled

NEI 99-01 Basis:

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of ALL AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment.

When in the Cold Shutdown, Refueling, or Defueled mode, this condition is not classified as an Alert because of the increased time available to restore another power source to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an essential bus. Some examples of this condition are presented below.

- A loss of ALL offsite power with a concurrent failure of ALL but one essential~~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 essential~~emergency~~ power sources (e.g., onsite diesel generators) with a single train of ~~emergency~~ essential buses being ~~back-~~fed from an offsite power source

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

The subsequent loss of the remaining single power source would escalate the event to an Alert in accordance with IC CA2.

DBNPS Basis:

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses C1 and D1 (ref. 1).

The essential buses during plant operation are normally powered from the 13.8KV offsite power system through their respective 13.8KV/4160V bus tie transformers, via the Unit Auxiliary Transformer (X11). In cold operating modes, the essential buses may be back-fed via the X11 and Main Transformer provided the main generator lead disconnect links are removed (ref. 1, 2). Credit for the X11 back-feed can only be taken if already aligned, as it takes greater than 15 minutes to align.

A standby source of offsite power to each 4160V essential bus is provided from the 13.8KV offsite power system via two separate and independent 13.8KV/4160V Startup Transformers (X01 & X02). Normally each startup transformer serves as a reserve power source for only one essential bus. However, a single startup transformer can be aligned to power both essential buses (ref. 1, 2).

Each essential bus has a diesel generator (EDG1 & EDG2) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power (ref. 1).

An Alternate AC power source, the SBO Diesel Generator, is located onsite. This onsite AC power source can be started from the Control Room and be loaded within 10 minutes of a SBO (ref. 1).

This cold condition EAL is equivalent to the hot condition EAL SA1.1.

DBNPS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. DB-OP-02521 Loss of AC Bus Power Sources
3. NEI 99-01 CU2

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-3 duration (Note 1, 10)

OR

UNPLANNED RCS pressure increase > 10 psig due to a loss of RCS cooling (This EAL does not apply during water-solid plant conditions).

Note 1: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

Note: 10 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.

Table C-3: RCS Heat-up Duration Thresholds

RCS Status	Containment Closure Status	Heat-up Duration
Intact (but **not** reduced inventory)	N/A	60 min.*
Not intact **OR** At reduced inventory	established	20 min.*
	not established	0 min.

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable.

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

NEI 99-01 Basis:

This IC addresses conditions involving a loss of decay heat removal capability or an addition of heat to the RCS in excess of that which can currently be removed. Either condition represents an actual or potential substantial degradation of the level of safety of the plant.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

The RCS Heat-up Duration Thresholds table addresses an increase in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not intact, or RCS inventory is reduced (e.g., mid-loop operation~~ in PWRs~~). The 20-minute criterion was included to allow time for operator action to address the temperature increase.

The RCS Heat-up Duration Thresholds table also addresses an increase in RCS temperature with the RCS INTACT. The status of CONTAINMENT CLOSURE is not crucial in this condition since the intact RCS is providing a high pressure barrier to a fission product release. The 60-minute time frame should allow sufficient time to address the temperature increase without a substantial degradation in plant safety.

Finally, in the case where there is an increase in RCS temperature, the RCS is not intact or is at reduced inventory~~ [*PWR*]~~, and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because 1) the evaporated reactor coolant may be released directly into the containment atmosphere and subsequently to the environment, and 2) there is reduced reactor coolant inventory above the top of irradiated fuel.

~~EAL #2~~The RCS pressure increase threshold provides a pressure-based indication of RCS heat-up in the absence of RCS temperature monitoring capability.

Escalation of the emergency classification level would be via IC CS1 or ~~AS1~~RS1.

DBNPS Basis:

In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the RCS pressure 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.

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1) including:

- Selected Incore thermocouples
- TIRC4B2 Reactor Coolant T-Cold Wide Range (Loop 1)
- TIRC4A2 Reactor Coolant T-Cold Wide Range (Loop 2)
- TIRC3B5 or TIRCB6 Reactor Coolant T-Hot (Loop 1)
- TIRC3A5 or TIRCA6 Reactor Coolant T-Hot (Loop 2)
- DHR Display on SPDS

(ref. 2)

A 10 psig RCS pressure increase can be read on various instruments such as (ref. 2):

- PI 6365B, Loop 1 Pressure

- PI 6365A, Loop 2 Pressure
- PRS RC2A1, Loop 2 Wide Range Pressure
- PI RC2A6, Low Range Pressure

The RCS is considered in reduced inventory if (ref. 3):

- The Reactor Coolant System is not full (loops not filled and the RCS is incapable of natural circulation), OR
- Both steam generators are not available as a heat sink, OR
- The refueling canal level is less than a stable level specified in the Shutdown Defense in Depth Review, typically 23 feet, with the head removed.

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications Table 1.1-1
2. DB-OP-02527 Loss of Decay Heat Removal
3. NG-DB-00117 Shutdown Defense In Depth Assessment
4. NEI 99-01 CA3

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 due to loss of decay heat removal capability	

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

NEI 99-01 Basis:

This IC addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit ~~, or the inability to determine RCS temperature and level,~~and represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Director should also refer to IC CA3.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

~~EAL #1~~This EAL involves a loss of decay heat removal capability, or an addition of heat to the ~~RCS~~ RCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

During an outage, the level in the reactor vessel will normally be maintained at or above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid increase in reactor coolant temperature depending on the time after shutdown.

~~EAL #2 reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.~~

~~Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.~~

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

DBNPS Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1) including:

- Selected Incore thermocouples
- TIRC4B2 Reactor Coolant T-Cold Wide Range (Loop 1)
- TIRC4A2 Reactor Coolant T-Cold Wide Range (Loop 2)
- TIRC3B5 or TIRCB6 Reactor Coolant T-Hot (Loop 1)
- TIRC3A5 or TIRCA6 Reactor Coolant T-Hot (Loop 2)
- DHR Display on SPDS

(ref. 2)

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.

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications Table 1.1-1 Modes
2. DB-OP-02527 Loss of Decay Heat Removal
3. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.2	**Unusual Event**
Loss of **ALL** RCS temperature and RCS level indication for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director 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

NEI 99-01 Basis:

This ~~IC~~ EAL addresses ~~an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit, or~~ the inability to determine RCS temperature and level, and represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Director should also refer to IC CA3.

~~A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.~~

~~EAL #1 involves a loss of decay heat removal capability, or an addition of heat to the RCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.~~

~~During an outage, the level in the reactor vessel will normally be maintained above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid increase in reactor coolant temperature depending on the time after shutdown.~~

~~EAL #2~~This EAL reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

DBNPS Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1) including:

- Selected Incore thermocouples
- TIRC4B2 Reactor Coolant T-Cold Wide Range (Loop 1)
- TIRC4A2 Reactor Coolant T-Cold Wide Range (Loop 2)
- TIRC3B5 or TIRCB6 Reactor Coolant T-Hot (Loop 1)
- TIRC3A5 or TIRCA6 Reactor Coolant T-Hot (Loop 2)
- DHR Display on SPDS

(ref. 2)

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 RCS level may be monitored by different means, including the ability to monitor level visually.

RCS level in the Refueling mode is normally monitored using the following instruments:

- LI 10596
- LI10577 A and B
- Clear tubing used for manometer level indication at the RCS cold legs.

RCS level indications (LI 10596, LI 10577A and B and cold leg tubing) provide accurate indication of water level when the RCS is at atmospheric pressure. However, there is no redundant means of RCS level indication when Mode 6 is entered (ref. 2).

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications Table 1.1-1
2. DB-OP-02527 Loss of Decay Heat Removal
3. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – Loss of Essential DC Power

Initiating Condition: Loss of ~~vital~~ essential DC power for 15 minutes or longer

EAL:

CU4.1	**Unusual Event**
< 105 VDC voltage indications on Technical Specification **required** essential 125 VDC distribution panels for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director 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

NEI 99-01 Basis

This IC addresses a loss of essential ~~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 an essential ~~vital~~ DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, "required" means the ~~vital~~ essential 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 essential ~~Vital~~ DC power affecting Train B would require the declaration of an Unusual Event. A loss of essential ~~Vital~~ DC power to Train A would not warrant an emergency classification.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Depending upon the event, escalation of the emergency classification level would be via IC CA1 or CA3, or an IC in Recognition Category ~~A~~R.

DBNPS Basis:

The DC electrical distribution subsystem consists of two trains, designated Train 1 and Train 2. Each train consists of a 250/125 VDC motor control center (MCC), and each 250/125 VDC MCC consists of two 125 VDC buses (one positive and one negative) that can be powered from a polarity specific battery or battery charger. The two DC buses then supply a specific Essential DC Distribution Panel (D1P and D1N for DCMCC 1, and D2P and D2N for DCMCC2). The four Essential DC Distribution Panels independently supply DC electrical

power to the four inverters, which in turn power the 120 VAC essential buses (ref. 1, 2, 3, 4, 5, 6).

The Class 1E DC loads have an operating voltage range of 105 to 135 volts. The minimum battery discharge voltage (requiring opening the degraded battery output breaker) is 105 VDC (ref. 1, 6).

This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS2.1.

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 8.0 Electrical Power
2. DB-OP-02537 Loss of D1P and DAP
3. DB-OP-02538 Loss of D2P and DBP
4. DB-OP-02539 Loss of D1N and DAN
5. DB-OP-02540 Loss of D2N and DBN
6. System Description for 125/250 VDC and 120 V Instrumentation AC System
7. 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-4 onsite communication methods

Table C-4 Communication Methods

System	Onsite	ORO	NRC
Public Address (Gaitronics)	X		
Onsite Radios	X		
Plant Telephones	X	X	X
Commercial Telephones	X	X	X
4-Way Ringdown Circuit		X	
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System (ETS)			X

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, D – Defueled

NEI 99-01 Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

~~EAL #1~~This EAL addresses a total loss of the communications methods used in support of routine plant operations.

DBNPS Basis:

Onsite/Offsite Response Organization (ORO)/NRC communications include one or more of the systems listed in Table C-4 (ref. 1, 2).

Public Address (Gaitronics) System

The DBNPS public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

On-site Radio System

Radio systems can be used for communication among operators, off-site monitoring teams, the control room, security, TSC and EOF.

Plant Telephone System

The DBNPS plant telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code as well as external or offsite calling capability.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by DBNPS. The local service provider provides primary and secondary power for their lines at the Central Office.

4-Way Ringdown Circuit

Dedicated ring down line that includes the State and County EOCs, the Ohio Highway Patrol Office, the Lucas County and Ottawa County Sheriff's dispatcher offices, the Emergency Operations Facility, and the Control Room.

Satellite Phones

Portable satellite telephones are available which enable communication when all other phone systems are inoperable, e.g. following a major external event. Internal batteries, external DC sources as well as external AC sources can power these portable systems.

Cellular Phones

Cellular phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telephone System

The NRC uses a DBNPS dedicated telephone line, which allows direct telephone communications from the plant to NRC regional and national offices. The DBNPS communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the DBNPS Control Room, Technical Support Center, and Emergency Operations Facility and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

This EAL is the cold condition equivalent of the hot condition EAL SU7.1.~~EAL #2 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).~~

~~EAL #3 addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.~~

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 9.5.2 Communications Systems
2. DBNPS Emergency Plan 7.6 Communications Systems
3. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

Initiating Condition: Loss of **ALL** onsite or offsite communications capabilities

EAL:

CU5.2 Unusual Event

Loss of **ALL** Table C-4 ORO communication methods

Table C-4 Communication Methods

System	Onsite	ORO	NRC
Public Address (Gaitronics)	X		
Onsite Radios	X		
Plant Telephones	X	X	X
Commercial Telephones	X	X	X
4-Way Ringdown Circuit		X	
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System (ETS)			X

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, D – Defueled

NEI 99-01 Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

~~EAL #1 addresses a total loss of the communications methods used in support of routine plant operations.~~

~~EAL #2~~This EAL 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, Lucas and Ottawa County EOCs.

DBNPS Basis:

Onsite/Offsite Response Organization (ORO)/NRC communications include one or more of the systems listed in Table C-4 (ref. 1, 2).

Public Address (Gaitronics) System

The DBNPS public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

On-site Radio System

Radio systems can be used for communication among operators, off-site monitoring teams, the control room, security, TSC and EOF.

Plant Telephone System

The DBNPS plant telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code as well as external or offsite calling capability.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by DBNPS. The local service provider provides primary and secondary power for their lines at the Central Office.

4-Way Ringdown Circuit

Dedicated ring down line that includes the State and County EOCs, the Ohio Highway Patrol Office, the Lucas County and Ottawa County Sheriff's dispatcher offices, the Emergency Operations Facility, and the Control Room.

Satellite Phones

Portable satellite telephones are available which enable communication when all other phone systems are inoperable, e.g. following a major external event. Internal batteries, external DC sources as well as external AC sources can power these portable systems.

Cellular Phones

Cellular phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telephone System

The NRC uses a DBNPS dedicated telephone line, which allows direct telephone communications from the plant to NRC regional and national offices. The DBNPS communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the DBNPS Control Room, Technical Support Center, and Emergency Operations Facility and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

This EAL is the cold condition equivalent of the hot condition EAL SU7.2.~~EAL #3 addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.~~

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 9.5.2 Communications Systems
2. DBNPS Emergency Plan 7.6 Communications Systems
3. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

Initiating Condition: Loss of **ALL** onsite or offsite communications capabilities

EAL:

CU5.3 Unusual Event

Loss of **ALL** Table C-4 NRC communication methods

Table C-4 Communication Methods

System	Onsite	ORO	NRC
Public Address (Gaitronics)	X		
Onsite Radios	X		
Plant Telephones	X	X	X
Commercial Telephones	X	X	X
4-Way Ringdown Circuit		X	
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System (ETS)			X

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, D – Defueled

NEI 99-01 Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

~~EAL #1 addresses a total loss of the communications methods used in support of routine plant operations.~~

~~EAL #2 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).~~

~~EAL #3~~This EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

DBNPS Basis:

Onsite/Offsite Response Organization (ORO)/NRC communications include one or more of the systems listed in Table C-4 (ref. 1, 2).

Public Address (Gaitronics) System

The DBNPS public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

On-site Radio System

Radio systems can be used for communication among operators, off-site monitoring teams, the control room, security, TSC and EOF.

Plant Telephone System

The DBNPS plant telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code as well as external or offsite calling capability.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by DBNPS. The local service provider provides primary and secondary power for their lines at the Central Office.

4-Way Ringdown Circuit

Dedicated ring down line that includes the State and County EOCs, the Ohio Highway Patrol Office, the Lucas County and Ottawa County Sheriff's dispatcher offices, the Emergency Operations Facility, and the Control Room.

Satellite Phones

Portable satellite telephones are available which enable communication when all other phone systems are inoperable, e.g. following a major external event. Internal batteries, external DC sources as well as external AC sources can power these portable systems.

Cellular Phones

Cellular phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telephone System

The NRC uses a DBNPS dedicated telephone line, which allows direct telephone communications from the plant to NRC regional and national offices. The DBNPS communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the DBNPS Control Room, Technical Support Center, and Emergency Operations Facility and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

This EAL is the cold condition equivalent of the hot condition EAL SU7.3.

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 9.5.2 Communications Systems
2. DBNPS Emergency Plan 7.6 Communications Systems
3. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting SAFETY SYSTEMS

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

EAL:

CA6.1 Alert

The occurrence of **ANY** Table C-5 Hazardous Event

AND EITHER:

- Event has caused indications of degraded performance in at least one train of a SAFETY SYSTEM required for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure required for the current operating mode

Table C-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Emergency Director

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

NEI 99-01 Basis:

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

~~EAL 1.b.1~~The first conditional addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

~~EAL 1.b.2~~The second conditional addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC CS1 or ~~AS1~~RS1.

DBNPS Basis:

- Ground motion acceleration of 0.08g horizontal or 0.053g vertical is the Maximum Probable Earthquake as is considered generically as the Operating Basis Earthquake for DBNPS (ref. 8). Control room alarm indication of an earthquake greater than OBE is indicated on the seismic control panel (C5764A). RA-EP-02820 Earthquake provides the guidance for determining any required response actions if the OBE earthquake threshold is exceeded (ref. 1). The significance of seismic events is discussed under EAL HU2.1.

- Internal FLOODING occurs from breaches of water systems that are located inside plant buildings and are connected to large water sources such as Intake Forebay or tanks (ref. 2).

- External FLOODING may be due to high lake level. DBNPS flood emergency elevation is 578 ft. Site access would be limited to rail, boat or helicopter (ref. 3).

- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 90 mph. (ref. 4).

- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area in the fire abnormal procedure (ref. 5).

- An EXPLOSION that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL (ref. 6).

DBNPS Basis Reference(s):

1. RA-EP-02820 Earthquake
2. RA-EP-02880 Internal Flooding
3. RA-EP-02830 Flooding
4. DBNPS UFSAR Section 3.3.1 Wind Criteria
5. DB-OP-02501 Serious Station Fire
6. RA-EP-02840 Explosion

7. NEI 99-01 CA6
8. Updated FSAR Section 3.1 Seismic Design

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 Essential AC Power

Loss of emergency electrical power can compromise plant SAFETY SYSTEM operability including decay heat removal and emergency core cooling systems, which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite sources for 4160 V essential buses.

2. Loss of Essential DC Power

Loss of emergency electrical power can compromise plant SAFETY SYSTEM operability including decay heat removal and emergency core cooling systems, which may be necessary to ensure fission product barrier integrity. This category includes loss of essential plant 125 VDC power sources.

3. Loss of Control Room Indications

Certain events that degrade plant operator's ability to assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant increase from these base-line levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

5. RCS Leakage

The reactor vessel provides a volume for the coolant that covers the reactor core. The reactor pressure vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad

integrity fail. Excessive RCS leakage greater than Technical Specification limits indicates potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and containment integrity.

6. RPS Failure

This subcategory includes events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RPS to complete a reactor trip comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean ANY trip failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and containment integrity.

7. Loss of Communications

Certain events that degrade plant operator's ability to communicate with essential personnel within or external to the plant warrant emergency classification.

8. Containment Failure

Failure of containment isolation capability (under conditions in which the containment is not currently challenged) warrants emergency classification. Failure of containment pressure control capability also warrants emergency classification.

9. Hazardous Event Affecting SAFETY SYSTEMS

Various natural and technological events that result in degraded plant SAFETY SYSTEM performance or significant visible damage warrant emergency classification under this subcategory.

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Prolonged loss of **ALL** offsite and **ALL** onsite AC power to essential buses

EAL:

SG1.1 General Emergency

Loss of **ALL** offsite and **ALL** onsite AC power capability, Table S-1, to essential 4160V buses C1 and D1

AND EITHER:

- Restoration of at least one essential bus in < 4 hours is **not** likely (Note 1)
- Calculated Clad Temperature in **Region 3** (DB-OP-02000 Figure 2)

Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 Offsite/Onsite AC Power Sources
Offsite: • X11 • X11 (back-fed via Main Transformer) • X01 • X02 **Onsite:** • EDG1 • EDG2 • SBODG

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

NEI 99-01 Basis:

This IC addresses a prolonged loss of ALL power sources to AC essential~~emergency~~ buses. A loss of ALL AC power compromises the performance of ALL SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A prolonged

loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

The EAL should require declaration of a General Emergency prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC essential~~emergency~~ bus by the end of the analyzed station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of challenges to multiple fission product barriers.

The estimate for restoring at least one essential~~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.

DBNPS Basis:

This EAL is indicated by the extended loss of ALL offsite and onsite AC power capability to 4160V essential buses C1 and D1 either for greater than 4 hours or that has resulted in indications of an actual loss of adequate core cooling.

Indication of continuing core cooling degradation is manifested by Calculated Clad Temperature in Region 3 (DB-OP-02000 Figure 2) (ref. 3).

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses C1 and D1 (ref. 1).

The essential buses during plant operation are normally powered from the 13.8KV offsite power system through their respective 13.8KV/4160V bus tie transformers, via the Unit Auxiliary Transformer (X11). In non-power operating modes, the essential buses may be back-fed via the X11 and Main Transformer provided the main generator lead disconnect links are removed (ref. 1, 2).

A standby source of offsite power to each 4160V essential bus is provided from the 13.8KV offsite power system via two separate and independent 13.8KV/4160V Startup Transformers (X01 & X02). Normally each startup transformer serves as a reserve power source for only one essential bus. However, a single startup transformer can be aligned to power both essential buses (ref. 1, 2).

Each essential bus has a diesel generator (EDG1 & EDG2) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power (ref. 1).

An Alternate AC power source, the Station Black Out (SBO) Diesel Generator, is located onsite. This onsite AC power source can be started from the Control Room and be loaded within 10 minutes of a SBO (ref. 1).

The SBODG fuel oil supply is separate from fuel oil supply for the station's EDGs. The SBODG fuel oil supply tank capacity is based on an eight-hour supply with the SBODG at rated load. A minimum supply of four hours must be stored in this tank to meet the site's station blackout duration analysis conditions (ref 1).

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on Emergency Director judgment as it relates to IMMINENT Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers. Indication of continuing core cooling degradation is manifested by Calculated Clad Temperature in Region 3 (DB-OP-02000 Figure 2). Figure 2, Incore T/C Temperature vs. RCS Pressure for ICC, provides indication of how serious core conditions are based upon combinations of RCS pressure and incore thermocouple temperatures. If the RCS P-T point is in Region 3, the cladding temperatures in the high power regions of the Core may be 1400ºF or higher. This is a serious condition, which could lead to significant amounts of H_2 production; Core damage may be unavoidable (ref. 3, 4).

DBNPS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. DB-OP-02521 Loss of AC Bus Power Sources
3. DB-OP-02000 Figure 2, Incore T/C Temperature vs. RC Pressure for Inadequate Core Cooling
4. Bases and Deviation Document for DB-OP-02000
5. NEI 99-01 SG1

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of **ALL** essential AC and DC power sources for 15 minutes or longer

EAL:

SG1.2 General Emergency

Loss of **ALL** offsite and **ALL** onsite AC power capability, Table S-1, to essential 4160V buses C1 and D1 for ≥ 15 min.

AND

Loss of **ALL** 125 VDC power based on battery bus voltage indications < 105 VDC on **ALL** essential DC distribution panels D1P, D1N, D2P and D2N for ≥ 15 min.

(Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 Offsite/Onsite AC Power Sources
Offsite: • X11 • X11 (back-fed via Main Transformer) • X01 • X02 **Onsite:** • EDG1 • EDG2 • SBODG

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

NEI 99-01 Basis:

This IC addresses a concurrent and prolonged loss of both essential AC AND ~~Vital~~ DC power. A loss of ALL essential 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 essential ~~vital~~ DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both AC and 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.

DBNPS Basis:

This EAL is indicated by the loss of ALL offsite and onsite essential AC power capability to 4160V essential buses C1 and D1 for greater than 15 minutes in combination with degraded DC power voltage. This EAL addresses operating experience from the March 2011 accident at Fukushima Daiichi.

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses C1 and D1 (ref. 1).

The essential buses during plant operation are normally powered from the 13.8KV offsite power system through their respective 13.8KV/4160V bus tie transformers, via the Unit Auxiliary Transformer (X11). In non-power operating modes, the essential buses may be back-fed via the X11 and Main Transformer provided the main generator lead disconnect links are removed (ref. 1, 2). Credit for the X11 back-feed can only be taken if already aligned, as it takes greater than 15 minutes to align.

A standby source of offsite power to each 4160V essential bus is provided from the 13.8KV offsite power system via two separate and independent 13.8KV/4160V Startup Transformers (X01 & X02). Normally each startup transformer serves as a reserve power source for only one essential bus. However, a single startup transformer can be aligned to power both essential buses (ref. 1, 2).

Each essential bus has a diesel generator (EDG1 & EDG2) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power (ref. 1).

An Alternate AC power source, the Station Black Out (SBO) Diesel Generator, is located onsite. This onsite AC power source can be started from the Control Room and be loaded within 10 minutes of a SBO (ref. 1).

The DC electrical distribution subsystem consists of two trains, designated Train 1 and Train 2. Each train consists of a 250/125 VDC motor control center (MCC), and each 250/125 VDC MCC consists of two 125 VDC buses (one positive and one negative) that can be powered

from a polarity specific battery or battery charger. The two DC buses then supply a specific Essential DC Distribution Panel (D1P and D1N for DCMCC 1, and D2P and D2N for DCMCC 2). The four Essential DC Distribution Panels independently supply DC electrical power to the four inverters, which in turn power the 120 VAC essential buses (ref. 1, 3, 4, 5, 6, 7).

The Class 1E DC loads have an operating voltage range of 105 to 135 volts. The minimum battery discharge voltage (requiring opening the degraded battery output breaker) is 105 VDC (ref. 1, 7).

DBNPS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. DB-OP-02521 Loss of AC Bus Power Sources
3. DB-OP-02537 Loss of D1P and DAP
4. DB-OP-02538 Loss of D2P and DBP
5. DB-OP-02539 Loss of D1N and DAN
6. DB-OP-02540 Loss of D2N and DBN
7. System Description for 125/250 VDC and 120 V Instrumentation AC System
8. NEI 99-01 SG8

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of **ALL** offsite power and **ALL** onsite AC power to essential 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 essential 4160V buses C1 and D1 for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 Offsite/Onsite AC Power Sources
Offsite: • X11 • X11 (back-fed via Main Transformer) • X01 • X02 **Onsite:** • EDG1 • EDG2 • SBODG

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

NEI 99-01 Basis:

This IC addresses a total loss of AC power that compromises the performance of ALL SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via ICs ~~AG1~~RG1, FG1 or SG1.

DBNPS Basis:

This EAL is indicated by the loss of ALL offsite and onsite AC power capability (Table S-1) to 4160V essential buses C1 and D1. The essential switchgear are buses C1 (Train A) and D1 (Train B) (ref. 1). For emergency classification purposes, “capability” means that an AC power source is available to the essential buses, whether or not the buses are powered from it.

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses C1 and D1 (ref. 1).

The essential buses during plant operation are normally powered from the 13.8KV offsite power system through their respective 13.8KV/4160V bus tie transformers, via the Unit Auxiliary Transformer (X11). In non-power operating modes, the essential buses may be back-fed via the X11 and Main Transformer provided the main generator lead disconnect links are removed (ref. 1, 2). Credit for the X11 back-feed can only be taken if already aligned, as it takes greater than 15 minutes to align.

A standby source of offsite power to each 4160V essential bus is provided from the 13.8KV offsite power system via two separate and independent 13.8KV/4160V Startup Transformers (X01 & X02). Normally each startup transformer serves as a reserve power source for only one essential bus. However, a single startup transformer can be aligned to power both essential buses (ref. 1, 2).

Each essential bus has a diesel generator (EDG1 & EDG2) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power (ref. 1).

An Alternate AC power source, the Station Black Out (SBO) Diesel Generator, is located onsite. This onsite AC power source can be started from the Control Room and be loaded within 10 minutes of a SBO (ref. 1).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. The interval begins when both offsite and onsite AC power capability are lost.

DBNPS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. DB-OP-02521 Loss of AC Bus Power Sources
3. NEI 99-01 SS1

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of **ALL** but one AC power source to essential buses for 15 minutes or longer

EAL:

SA1.1 Alert

AC power capability, Table S-1, to essential 4160V buses C1 and D1 reduced to a single power source for ≥ 15 min. (Note 1)

AND

ANY additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS

Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 Offsite/Onsite AC Power Sources
Offsite: • X11 • X11 (back-fed via Main Transformer) • X01 • X02 **Onsite:** • EDG1 • EDG2 • SBODG

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4- Hot Shutdown

Basis:

NEI 99-01 Basis:

This IC describes a significant degradation of offsite and onsite AC power sources such that ANY additional single failure would result in a loss of ALL AC power to SAFETY SYSTEMS.

In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

An “AC power source” is a source recognized in AOPs and EOPs, and capable of supplying required power to an essential~~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 essential~~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 essential~~emergency~~ buses being ~~back-~~fed from an offsite power source

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

Escalation of the emergency classification level would be via IC SS1.

DBNPS Basis:

For emergency classification purposes, “capability” means that an AC power source is available to the essential buses, whether or not the buses are powered from it.

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses C1 and D1 (ref. 1).

The essential buses during plant operation are normally powered from the 13.8KV offsite power system through their respective 13.8KV/4160V bus tie transformers, via the Unit Auxiliary Transformer (X11). In non-power operating modes, the essential buses may be back-fed via the X11 and Main Transformer provided the main generator lead disconnect links are removed (ref. 1, 2). Credit for the X11 back-feed can only be taken if already aligned, as it takes greater than 15 minutes to align.

A standby source of offsite power to each 4160V essential bus is provided from the 13.8KV offsite power system via two separate and independent 13.8KV/4160V Startup Transformers (X01 & X02). Normally each startup transformer serves as a reserve power source for only one essential bus. However, a single startup transformer can be aligned to power both essential buses (ref. 1, 2).

Each essential bus has a diesel generator (EDG1 & EDG2) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power (ref. 1).

An Alternate AC power source, the Station Black Out (SBO) Diesel Generator, is located onsite. This onsite AC power source can be started from the Control Room and be loaded within 10 minutes of a SBO (ref. 1).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. If the capability of a second source of essential bus power is not restored within 15 minutes, an Alert is declared under this EAL.

DBNPS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. DB-OP-02521 Loss of AC Bus Power Sources
3. NEI 99-01 SA1

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of **ALL** offsite AC power capability to essential buses for 15 minutes or longer

EAL:

SU1.1 Unusual Event

Loss of **ALL** offsite AC power capability, Table S-1, to essential 4160V buses C1 and D1 for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 Offsite/Onsite AC Power Sources
Offsite: • X11 • X11 (back-fed via Main Transformer) • X01 • X02 **Onsite:** • EDG1 • EDG2 • SBODG

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Hot Shutdown

Basis:

NEI 99-01 Basis:

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC essential~~emergency~~ buses. This condition represents a potential reduction in the level of safety of the plant.

For emergency classification purposes, “capability” means that an offsite AC power source(s) is available to the essential~~emergency~~ buses, whether or not the buses are powered from it.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of offsite power.

Escalation of the emergency classification level would be via IC SA1.

DBNPS Basis:

For emergency classification purposes, “capability” means that an AC power source is available to the essential buses, whether or not the buses are powered from it.

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses C1 and D1 (ref. 1).

The essential buses during plant operation are normally powered from the 13.8KV offsite power system through their respective 13.8KV/4160V bus tie transformers, via the Unit Auxiliary Transformer (X11). In non-power operating modes, the essential buses may be back-fed via the X11 and Main Transformer provided the main generator lead disconnect links are removed (ref. 1, 2). Credit for the X11 back-feed can only be taken if already aligned, as it takes greater than 15 minutes to align.

A standby source of offsite power to each 4160V essential bus is provided from the 13.8KV offsite power system via two separate and independent 13.8KV/4160V Startup Transformers (X01 & X02). Normally each startup transformer serves as a reserve power source for only one essential bus. However, a single startup transformer can be aligned to power both essential buses (ref. 1, 2).

Each essential bus has a diesel generator (EDG1 & EDG2) to supply an onsite emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of off-site power (ref. 1).

An Alternate AC power source, the Station Black Out (SBO) Diesel Generator, is located onsite. This onsite AC power source can be started from the Control Room and be loaded within 10 minutes of a SBO (ref. 1).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses.

DBNPS Basis Reference(s):

1. UFSAR Section 8.0 Electric Power
2. DB-OP-02521 Loss of AC Bus Power Sources
3. NEI 99-01 SU1

Category: S – System Malfunction

Subcategory: 2 – Loss of Essential DC Power

Initiating Condition: Loss of **ALL** essential 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** essential DC distribution panels D1P, D1N, D2P and D2N for ≥ 15 min. (Note 1)

Note 1: The Emergency Director 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

NEI 99-01 Basis:

This IC addresses a loss of ~~vital~~ essential 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.

DBNPS Basis:

The DC electrical distribution subsystem consists of two trains, designated Train 1 and Train 2. Each train consists of a 250/125 VDC motor control center (MCC), and each 250/125 VDC MCC consists of two 125 VDC buses (one positive and one negative) that can be powered from a polarity specific battery or battery charger. The two DC buses then supply a specific Essential DC Distribution Panel (D1P and D1N for DCMCC 1, and D2P and D2N for DCMCC 2). The four Essential DC Distribution Panels independently supply DC electrical power to the four inverters, which in turn power the 120 VAC ~~vital~~ essential buses (ref. 1, 2, 3, 4, 5, 6).

The Class 1E DC loads have an operating voltage range of 105 to 135 volts. The minimum battery discharge voltage (requiring opening the degraded battery output breaker) is 105 VDC (ref. 1, 6).

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 8.0 Electrical Power
2. DB-OP-02537 Loss of D1P and DAP
3. DB-OP-02538 Loss of D2P and DBP
4. DB-OP-02539 Loss of D1N and DAN
5. DB-OP-02540 Loss of D2N and DBN
6. System Description for 125/250 VDC and 120 V Instrumentation AC System
7. NEI 99-01 SS8

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer 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 SAFETY SYSTEM parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

ANY Significant Transient is or may be in progress, Table S-3

Note 1: The Emergency Director 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 pressure • In-core T/C temperature • Level in at least one S/G • Auxiliary or emergency feed flow

Table S-3 Significant Transients
• Reactor trip • Runback > 25% thermal power • Electrical load rejection > 25% electrical load • Safety injection actuation

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

NEI 99-01 Basis:

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an “inability to monitor” means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of ALL of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling ~~[*PWR*] / RPV level [*BWR*]~~ and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if ALL indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for ~~reactor vessel~~RCS ~~level~~ pressure ~~[*PWR*] / RPV water level [*BWR*]~~ cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via ICs FS1 or ~~AS1~~RS1

DBNPS Basis:

SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Process Computer, which displays SPDS required information, serves as a redundant compensatory indicator, which may be utilized in lieu of normal Control Room indicators (ref. 1, 2, 3).

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 25% thermal power

change, electrical load rejections of greater than 25% full electrical load or SI injection actuations.

DBNPS Basis Reference(s):

1. UFSAR Section 7.5 Safety-Related Display Instrumentation
2. DB-OP-02541 Loss of YAU
3. DB-OP-02542 Loss of YBU
4. NEI 99-01 SA2

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 SAFETY SYSTEM parameters from within the Control Room for ≥ 15 min. (Note 1)

Note 1: The Emergency Director 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 pressure • In-core T/C temperature • Level in at least one S/G • Auxiliary or emergency feed flow

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

NEI 99-01 Basis:

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an “inability to monitor” means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of ALL of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating

procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling ~~[*PWR*] / RPV level [*BWR*]~~ and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if ALL indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for ~~reactor vessel~~RCS ~~level~~ pressure~~[PWR] / RPV water level [BWR]~~ cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via IC ~~SA2~~SA3.

DBNPS Basis:

SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Process Computer, which displays SPDS required information, serves as a redundant compensatory indicator, which may be utilized in lieu of normal Control Room indicators (ref. 1, 2, 3).

DBNPS Basis Reference(s):

1. UFSAR Section 7.5 Safety-Related Display Instrumentation
2. DB-OP-02541 Loss of YAU
3. DB-OP-02542 Loss of YBU
4. NEI 99-01 SU2

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.1	**Unusual Event**
Letdown Monitor (RE 1998) reading > 2.0E+06 cpm	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

NEI 99-01 Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category ~~A~~ R ICs.

DBNPS Basis:

The DBNPS historical highest failed fuel level of 0.007% corresponds to a reading of 1.4E+05 cpm on the Failed Fuel Monitor (RE 1998). The top of scale for RE 1998 is 1.0E+07. A monitor value of 2.0E+06 (0.1% clad damage) was chosen for its ability to be recognized even though exceeding Technical Specification Limits could potentially result in much higher readings (ref. 1, 2).

DBNPS Basis Reference(s):

1. Radiation Monitor Setpoint Manual, RE 1998 Failed Fuel Detector
2. DBNPS Technical Specifications LCO 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

RCS activity > Technical Specification LCO 3.4.16 as indicated by **ANY** of the following:

- Dose equivalent I-131 in the unacceptable region of Figure 3.4.16-1
- > 1 µCi/gm dose equivalent I-131 for > 48 hrs
- > 100/Ē µCi/gm gross specific coolant activity

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

NEI 99-01 Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category ~~A~~ R ICs.

DBNPS Basis:

Technical Specification LCO 3.4.16 Condition A limits RCS System Dose Equivalent I-131 to ≤ 1.0 µCi/gm for > 48 hours. Technical Specification Section 3.4.16 Condition B limits RCS System Dose Equivalent I-131 to within Figure 3.4.16-1 and gross specific activity to ≤ 100/Ē µCi/gm. (ref 1).

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications section 3.4.16 RCS Specific Activity
2. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1	**Unusual Event**
RCS leakage > 10 gpm for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director 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

NEI 99-01 Basis:

This IC addresses RCS leakage, which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

~~EAL #1 and EAL #2 are~~This EAL is focused on a loss of mass from the RCS due to "unidentified leakage" or "pressure boundary leakage" ~~or "identified leakage"~~ (as these leakage types are defined in the plant Technical Specifications). ~~EAL #3 addresses a RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These EALs thus apply to leakage into the containment, a secondary-side system (e.g., steam generator tube leakage in a PWR) or a location outside of containment.~~

The leak rate value~~s for each EAL were~~ was selected because ~~they are~~it is usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). ~~EAL #1 uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.~~

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. ~~For PWRs, a~~An emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated). ~~For BWRs, a stuck-open Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.~~

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

Escalation of the emergency classification level would be via ICs of Recognition Category ~~A~~ R or F.

DBNPS Basis:

Unidentified leakage is ALL leakage (except RCP seal return) that is not identified leakage (ref. 1, 2).

DBNPS does not have the capability to classify leakage as identified leakage within 15 minutes. Therefore, for the purpose of this IC and EAL all RCS leakage is considered unidentified leakage and the 10 gpm leak rate applies.

Escalation of this EAL to the Alert level is via Category F, Fission Product Barrier Degradation, EAL FA1.1.

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications Definitions section 1.1
2. UFSAR Section 5.2.4.7 Leakage Identification
3. DB-OP-02522 Small RCS Leaks
4. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%

AND

ALL actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5%

AND EITHER:

- Calculated Clad Temperature in **Region 3** (DB-OP-02000 Figure 2)
- MFW, AFW and MU-HPI PORV Cooling are **all** unavailable

Mode Applicability:

1 - Power Operation

NEI 99-01 Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(~~trip ~~[*PWR*] / scram [*BWR*])~~ that results in a reactor shutdown, ALL subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shut down the reactor. The inclusion of this IC and EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Escalation of the emergency classification level would be via ICs ~~AG1~~ RG1 or FG1.

DBNPS Basis:

This EAL addresses the following:

- ANY automatic reactor trip signal followed by a manual trip that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the SAFETY SYSTEMS were designed (EAL SA6.1), AND

- Indications that either core cooling is extremely challenged or heat removal is extremely challenged

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS barriers.

Reactor shutdown achieved by use of other trip actions specified in DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture (locally opening Reactor Trip Breaker, emergency boration or manually driving control rods) are also credited as a successful manual trip methods provided reactor power can be reduced below 5% before indications of an extreme challenge to either core cooling or heat removal exist (ref. 1).

Each of the redundant Auxiliary Feed Pumps is sized to provide 100% of the capacity required by the SGs to remove 5% of the reactor thermal power produced at full load steam pressure conditions. The ≥ 5% power portion of this EAL threshold was chosen to be an easily recognizable, onscale indication that the reactor trip has not functioned to shutdown the reactor assuming the reactor power was in the normal operating range.

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on Emergency Director judgment as it relates to IMMINENT Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers. Indication of continuing core cooling degradation is manifested by Calculated Clad Temperature in Region 3 (DB-OP-02000 Figure 2). Figure 2, Incore T/C Temperature vs. RCS Pressure for ICC, provides indication of how serious core conditions are based upon combinations of RCS pressure and incore thermocouple temperatures. If the RCS P-T point is in Region 3, the cladding temperatures in the high power regions of the core may be 1400°F or higher. This is a serious condition, which will lead to significant amounts of H2 production; core damage may be unavoidable (ref. 1).

An extreme challenge to heat removal is defined as a complete loss of MFW, AFW and MU-HPI PORV Cooling (ref. 2, 3).

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications section 3.3.1 Reactor Protection System (RPS) Instrumentation
2. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
3. Bases and Deviation Document for DB-OP-02000

4. NEI 99-01 SS5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor AND subsequent manual actions taken at the ~~reactor control consoles~~Controls Area 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 Controls Area (manual RPS trip pushbuttons and de-energizing E2 and F2) 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

NEI 99-01 Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(~~trip ~~[*PWR*] / scram [*BWR*])~~ that results in a reactor shutdown, AND subsequent operator manual actions taken at the ~~reactor control consoles~~Controls Area to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the ~~reactor control consoles~~Controls Area since this event entails a significant failure of the RPS.

A manual action at the ~~reactor~~ Controls ~~console~~ Area 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 consoles~~Controls Area (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~~ Controls ~~console~~Areas~~s~~".~~Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action. [*BWR*]~~

The plant response to the failure of an automatic or manual reactor ~~(~~trip ~~[*PWR*] / scram [*BWR*])~~ will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shut down the reactor is prolonged enough to cause a challenge to the core cooling~~[PWR] / RPV water level [BWR]~~ or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS6~~5~~. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6~~5~~ or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

DBNPS Basis:

This EAL addresses ANY automatic or manual reactor trip signal that fails to shut down the reactor 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.

For the purposes of emergency classification, successful manual trip actions are those, which can be quickly performed from the Controls Area (ATCA) (i.e., manual pushbuttons or de-energizing E2 and F2). Reactor shutdown achieved by use of other trip actions specified in DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture (locally opening Reactor Trip Breaker, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 1).

Each of the redundant Auxiliary Feed Pumps is sized to provide 100% of the capacity required by the SGs to remove 5% of the reactor thermal power produced at full load steam pressure conditions. The ≥ 5% power portion of this EAL threshold was chosen to be an easily recognizable, onscale indication that the reactor trip has not functioned to shutdown the reactor assuming the reactor power was in the normal operating range.

Escalation of this event to a Site Area Emergency would be under EAL SS6.1 or Emergency Director judgment.

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

SU6.1 Unusual Event

An automatic trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **ANY** RPS setpoint is exceeded

AND

A subsequent manual trip action taken at the Controls Area (manual RPS trip pushbuttons or de-energizing E2 and F2) 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

NEI 99-01 Basis:

This ~~IC~~ EAL addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(~~trip ~~[*PWR*] / scram [*BWR*])~~that results in a reactor shutdown, AND~~either~~ a subsequent operator manual action taken at the ~~reactor control consoles~~Controls Area ~~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~~Controls Area to shutdown the reactor (e.g., initiate a manual reactor ~~(~~trip~~[*PWR*] / scram [*BWR*])~~). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor ~~(~~trip~~[*PWR*] / scram [*BWR*])~~ is unsuccessful, operators will promptly take manual action at another location(s) on the ~~reactor control consoles~~Controls Area to shutdown the reactor (e.g., initiate a manual reactor ~~(~~trip~~[*PWR*] / scram [*BWR*])~~) using a different switch). Depending upon several factors, the initial or subsequent effort to manually ~~(~~trip ~~[*PWR*] / scram [*BWR*])~~the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor ~~(~~trip ~~[*PWR*] / scram [*BWR*])~~signal. If a subsequent manual or automatic ~~(~~trip ~~[*PWR*] / scram [*BWR*])~~is successful in shutting down the reactor, core heat

generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the ~~reactor~~ Controls ~~consoles~~ Area is ANY operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor ~~(~~trip~~[*PWR*] / scram [*BWR*])~~). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the ~~reactor control consoles~~Controls Area".~~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~~Controls Area are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC ~~SA5~~SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC ~~SA5~~ SA6 or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor ~~(~~trip ~~[PWR] / scram [BWR])~~signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor ~~(~~trip ~~[PWR] / scram [BWR])~~ and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.

- If the signal does not cause a plant transient and the ~~(~~trip ~~[PWR] / scram [BWR])~~failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

DBNPS Basis:

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) trip function. A reactor trip is automatically initiated by the RPS when certain continuously monitored parameters exceed predetermined setpoints (ref. 1).

Following a successful reactor trip, rapid insertion of the control rods occurs. Reactor power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from an automatic reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful trip has therefore occurred when

there is sufficient rod insertion from the trip of RPS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2).

Each of the redundant Auxiliary Feed Pumps is sized to provide 100% of the capacity required by the SGs to remove 5% of the reactor thermal power produced at full load steam pressure conditions. The ≥ 5% power portion of this EAL threshold was chosen to be an easily recognizable, onscale indication that the reactor trip has not functioned to shutdown the reactor assuming the reactor power was in the normal operating range.

For the purposes of emergency classification, successful manual trip actions are those, which can be quickly performed from the Controls Area (ATCA) (i.e., manual pushbuttons or de-energizing E2 and F2). Reactor shutdown achieved by use of other trip actions specified in DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture (locally opening Reactor Trip Breaker, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 2).

In the event that the operator identifies a reactor trip is imminent and initiates a successful manual reactor trip before the automatic RPS trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. However, if subsequent manual reactor trip actions fail to reduce reactor power below 5%, the event escalates to the Alert under EAL SA6.1.

If by procedure, operator actions include the initiation of an immediate manual trip following receipt of an automatic trip signal and there are no clear indications that the automatic trip failed (such as a time delay following indications that a trip setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic trip or manual actions. If a subsequent review of the trip actuation indications reveals that, the automatic trip did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 10 CFR 50.72 should be considered for the transient event.

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications section 3.3.1 Reactor Protection System (RPS) Instrumentation
2. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
3. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

SU6.2 Unusual Event

A manual trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **ANY** manual trip action was initiated

AND

A subsequent automatic trip or manual trip action taken at the Controls Area (manual RPS trip pushbuttons or de-energizing E2 and F2) 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

NEI 99-01 Basis:

This ~~IC~~ EAL addresses a failure of the RPS to initiate or complete a~~n 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~~Controls Area 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~~Controls Area to shutdown the reactor (e.g., initiate a manual reactor ~~(~~trip~~[*PWR*] / scram [*BWR*])~~). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor ~~(~~trip~~[*PWR*] / scram [*BWR*])~~ is unsuccessful, operators will promptly take manual action at another location(s) on the ~~reactor control consoles~~Controls Area to shutdown the reactor (e.g., initiate a manual reactor ~~(~~trip~~[*PWR*] / scram [*BWR*])~~) using a different switch~~)~~. Depending upon several factors, the initial or subsequent effort to manually trip ~~[*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~~ Controls ~~consoles~~ Area is ANY operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor ~~(~~trip~~[*PWR*] / scram [*BWR*])~~). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the ~~reactor control consoles~~Controls Area".

~~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~~Controls Area are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC ~~SA5~~SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC ~~SA5~~ SA6 or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor ~~(~~trip ~~[PWR] / scram [BWR])~~signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor ~~(~~trip ~~[PWR] / scram [BWR])~~ and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.

- If the signal does not cause a plant transient and the ~~(~~trip ~~[PWR] / scram [BWR])~~failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

DBNPS Basis:

This EAL addresses a failure of a manually initiated trip in the absence of having exceeded an automatic RPS trip setpoint AND a subsequent automatic or manual trip is successful in shutting down the reactor (reactor power < 5%) (ref. 1).

Following a successful reactor trip, rapid insertion of the control rods occurs. Reactor 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 RPS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2).

Each of the redundant Auxiliary Feed Pumps is sized to provide 100% of the capacity required by the SGs to remove 5% of the reactor thermal power produced at full load steam pressure conditions. The ≥ 5% power portion of this EAL threshold was chosen to be an easily recognizable, onscale indication that the reactor trip has not functioned to shutdown the reactor assuming the reactor power was in the normal operating range.

For the purposes of emergency classification, successful manual trip actions are those, which can be quickly performed from the Controls Area (ATCA) (i.e., manual pushbuttons or de-energizing E2 and F2). Reactor shutdown achieved by use of other trip actions specified in DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture (locally opening Reactor Trip Breaker, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 2).

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

DBNPS Basis Reference(s):

1. DBNPS Technical Specifications section 3.3.1 Reactor Protection System (RPS) Instrumentation
2. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
3. NEI 99-01 SU5

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

Table S-4 Communication Methods

System	Onsite	ORO	NRC
Public Address (Gaitronics)	X		
Onsite Radios	X		
Plant Telephones	X	X	X
Commercial Telephones	X	X	X
4-Way Ringdown Circuit		X	
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System			X

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

NEI 99-01 Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

~~EAL #1~~This EAL addresses a total loss of the communications methods used in support of routine plant operations.

DBNPS Basis:

Onsite/Offsite Response Organization (ORO)/NRC communications include one or more of the systems listed in Table S-4 (ref. 1, 2).

Public Address (Gaitronics) System

The DBNPS public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

On-site Radio System

Radio systems can be used for communication among operators, off-site monitoring teams, the control room, security, TSC and EOF.

Plant Telephone System

The DBNPS plant telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code as well as external or offsite calling capability.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by DBNPS. The local service provider provides primary and secondary power for their lines at the Central Office.

4-Way Ringdown Circuit

Dedicated ring down line that includes the State and County EOCs, the Ohio Highway Patrol Office, the Lucas County and Ottawa County Sheriff's dispatcher offices, the Emergency Operations Facility, and the Control Room.

Satellite Phones

Portable satellite telephones are available which enable communication when all other phone systems are inoperable, e.g. following a major external event. Internal batteries, external DC sources as well as external AC sources can power these portable systems.

Cellular Phones

Cellular phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telephone System

The NRC uses a DBNPS dedicated telephone line, which allows direct telephone communications from the plant to NRC regional and national offices. The DBNPS communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the DBNPS Control Room, Technical Support Center, and Emergency Operations Facility and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

~~EAL #2 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).~~

~~EAL #3 addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.~~

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 9.5.2 Communications Systems
2. DBNPS Emergency Plan 7.6 Communications Systems
3. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

Initiating Condition: Loss of **ALL** onsite or offsite communications capabilities

EAL:

SU7.2 Unusual Event

Loss of **ALL** Table S-4 ORO communication methods

Table S-4 Communication Methods

System	Onsite	ORO	NRC
Public Address (Gaitronics)	X		
Onsite Radios	X		
Plant Telephones	X	X	X
Commercial Telephones	X	X	X
4-Way Ringdown Circuit		X	
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System			X

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

NEI 99-01 Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

~~EAL #1 addresses a total loss of the communications methods used in support of routine plant operations.~~

~~EAL #2~~This EAL 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 of Ohio, Lucas and Ottawa County EOCs.

DBNPS Basis:

Onsite/Offsite Response Organization (ORO)/NRC communications include one or more of the systems listed in Table S-4 (ref. 1, 2).

Public Address (Gaitronics) System

The DBNPS public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

On-site Radio System

Radio systems can be used for communication among operators, off-site monitoring teams, the control room, security, TSC and EOF.

Plant Telephone System

The DBNPS plant telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code as well as external or offsite calling capability.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by DBNPS. The local service provider provides primary and secondary power for their lines at the Central Office.

4-Way Ringdown Circuit

Dedicated ring down line that includes the State and County EOCs, the Ohio Highway Patrol Office, the Lucas County and Ottawa County Sheriff's dispatcher offices, the Emergency Operations Facility, and the Control Room.

Satellite Phones

Portable satellite telephones are available which enable communication when ALL other phone systems are inoperable, e.g. following a major external event. Internal batteries, external DC sources as well as external AC sources can power these portable systems.

Cellular Phones

Cellular phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telephone System

The NRC uses a DBNPS dedicated telephone line, which allows direct telephone communications from the plant to NRC regional and national offices. The DBNPS communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the DBNPS Control Room, Technical Support Center, and Emergency Operations Facility and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

This EAL is the hot condition equivalent of the cold condition EAL CU5.2.

~~EAL #3 addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.~~

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 9.5.2 Communications Systems
2. DBNPS Emergency Plan 7.6 Communications Systems
3. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

Initiating Condition: Loss of **ALL** onsite or offsite communications capabilities

EAL:

SU7.3 Unusual Event

Loss of **ALL** Table S-4 NRC communication methods

Table S-4 Communication Methods

System	Onsite	ORO	NRC
Public Address (Gaitronics)	X		
Onsite Radios	X		
Plant Telephones	X	X	X
Commercial Telephones	X	X	X
4-Way Ringdown Circuit		X	
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System			X

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

NEI 99-01 Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

~~EAL #1 addresses a total loss of the communications methods used in support of routine plant operations.~~

~~EAL #2 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).~~

~~EAL #3~~This EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

DBNPS Basis:

Onsite/Offsite Response Organization (ORO)/NRC communications include one or more of the systems listed in Table S-4 (ref. 1, 2).

Public Address (Gaitronics) System

The DBNPS public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

On-site Radio System

Radio systems can be used for communication among operators, off-site monitoring teams, the control room, security, TSC and EOF.

Plant Telephone System

The DBNPS plant telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code as well as external or offsite calling capability.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by DBNPS. The local service provider provides primary and secondary power for their lines at the Central Office.

4-Way Ringdown Circuit

Dedicated ring down line that includes the State and County EOCs, the Ohio Highway Patrol Office, the Lucas County and Ottawa County Sheriff's dispatcher offices, the Emergency Operations Facility, and the Control Room.

Satellite Phones

Portable satellite telephones are available which enable communication when ALL other phone systems are inoperable, e.g. following a major external event. Internal batteries, external DC sources as well as external AC sources can power these portable systems.

Cellular Phones

Cellular phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telephone System

The NRC uses a DBNPS dedicated telephone line, which allows direct telephone communications from the plant to NRC regional and national offices. The DBNPS communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the DBNPS Control Room, Technical Support Center, and Emergency Operations Facility and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

This EAL is the hot condition equivalent of the cold condition EAL CU5.3.

DBNPS Basis Reference(s):

1. DBNPS UFSAR Section 9.5.2 Communications Systems
2. DBNPS Emergency Plan 7.6 Communications Systems
3. 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 closed within 15 min. of a VALID containment isolation signal (Note 1)	

Note 1: The Emergency Director 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

NEI 99-01 Basis:

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~~ this condition represents potential degradation of the level of safety of the plant.

For ~~EAL #1,~~this EAL 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. 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 addresses a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment sprays or ice condenser 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.

DBNPS Basis:

Successful closure of any one valve in a penetration line is sufficient to consider the penetration closed.

DBNPS Basis Reference(s):

1. NEI 99-01 SU7

Category: S – System Malfunction

Subcategory: 8 – Containment Failure

Initiating Condition: Failure to isolate containment or loss of containment pressure control

EAL:

SU8.2 Unusual Event

Containment pressure > 40 psia with < one full train of containment cooling, Table S-6, operating per design for ≥ 15 min. (Note 1)

Table S-6 Containment Cooling Full Train	
CT Spray Pumps	**CT Cooling Fans**
2	0
1	1
0	2

Note 1: The Emergency Director 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

NEI 99-01 Basis:

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~~ this condition represents a potential degradation of the level of safety of the plant.

~~For EAL #1, 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. 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~~This EAL addresses a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not

have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment sprays or ice condenser 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.

DBNPS Basis:

The combination of Containment spray pumps and Containment cooling fan units considered to be a full train of containment cooling operating per design is shown in Table F-3 (ref. 1).

SFAS 2 actuation automatically initiates Containment Air Coolers upon exceeding the Containment pressure high setpoint of 18.7 psia or low RCS pressure of 1600 psig. SFAS Level 4 actuation automatically initiates Containment Spray upon exceeding the Containment pressure high-high setpoint of 40 (nominal) psia (ref. 2, 3).

DBNPS Basis Reference(s):

1. UFSAR Section 6.2.2. Containment Vessel Heat Removal Systems
2. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
3. DBNPS Technical Specifications Table 3.3.5-1 Safety Features Actuation System Instrumentation
4. 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 has caused indications of degraded performance in at least one train of a SAFETY SYSTEM required for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure required for the current operating mode

Table S-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Emergency Director

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

NEI 99-01 Basis:

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

~~EAL 1.b.1~~The first conditional addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

~~EAL 1.b.2~~The second conditional addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC FS1 or ~~AS1~~RS1.

DBNPS Basis:

- Ground motion acceleration of 0.08g horizontal or 0.053g vertical is the Maximum Probable Earthquake as is considered generically as the Operating Basis Earthquake for DBNPS (ref. 8). Control room alarm indication of an earthquake greater than OBE is indicated on the seismic control panel (C5764A). RA-EP-02820 Earthquake provides the guidance for determining any required response actions if the OBE earthquake threshold is exceeded (ref. 1). The significance of seismic events is discussed under EAL HU2.1.

- Internal FLOODING occurs from breaches of water systems that are located inside plant buildings and are connected to large water sources such as Intake Forebay or tanks (ref. 2).

- External FLOODING may be due to high lake level. DBNPS flood emergency elevation is 578 ft. Site access would be limited to rail, boat or helicopter (ref. 3).

- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 90 mph. (ref. 4).

- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area in the fire abnormal procedure (ref. 5).

- An EXPLOSION that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL (ref. 6).

DBNPS Basis Reference(s):

1. RA-EP-02820 Earthquake
2. RA-EP-02880 Internal Flooding
3. RA-EP-02830 Flooding
4. DBNPS UFSAR Section 3.3.1 Wind Criteria
5. DB-OP-02501 Serious Station Fire
6. RA-EP-02840 Explosion
7. NEI 99-01 SA9
8. Updated FSAR Section 3.1 Seismic Design

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 (CT): The Containment Barrier includes the containment building, connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 2). “Loss” and “Potential Loss” signify the relative damage and threat of damage to the barrier. “Loss” means the barrier no longer assures containment of radioactive materials. “Potential Loss” means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Alert:

ANY loss or ANY potential loss of EITHER Fuel Clad or RCS

Site Area Emergency:

Loss or potential loss of ANY two barriers

General Emergency:

Loss of ANY two barriers AND loss or potential loss of third barrier

The logic used for emergency classification based on fission product barrier monitoring should reflect the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier.

- Unusual Event ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.

- For accident conditions involving a radiological release, evaluation of the FISSION PRODUCT BARRIER THRESHOLDS will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the FISSION PRODUCT BARRIER THRESHOLDS may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC RG1 has been exceeded.

- The FISSION PRODUCT BARRIER THRESHOLDS specified within a scheme reflect plant-specific DBNPS design and operating characteristics.

- As used in this category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside the containment, an interfacing system, or outside of the containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered RCS leakage.

- At the Site Area Emergency level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: 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

NEI 99-01 Basis:

None

DBNPS 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

DBNPS Basis Reference(s):

1. NEI 99-01 FG1

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

NEI 99-01 Basis:

None

DBNPS 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 assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, the Emergency Director would have greater assurance that escalation to a General Emergency is less IMMINENT.

DBNPS Basis Reference(s):

1. NEI 99-01 FS1

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

NEI 99-01 Basis:

None

DBNPS 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

DBNPS Basis Reference(s):

1. NEI 99-01 FA1

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. CT Radiation / RCS Activity

D. CT Integrity or Bypass

E. Emergency Director Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more FISSION PRODUCT BARRIER THRESHOLDS appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word “None” is entered in the cell.

Thresholds are assigned sequential numbers within each 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 “CT 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 (CT) 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 (SFAS) actuation required by **EITHER**: • UNISOLABLE RCS leakage • SG tube RUPTURE	1. Operation of a standby Makeup Pump (>250 gpm) is required by **EITHER:** • UNISOLABLE RCS leakage • SG tube leakage **OR** 2. PTS requirements invoked (SR5)	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B **Inadequate Heat Removal**	1. Calculated Clad Temperature in **Region 3** or higher (DB-OP-02000 Figure 2)	1. Calculated Clad Temperature in **Region 2** or higher (DB-OP-02000 Figure 2) **OR** 2. Loss of **ALL** feedwater **AND** SG Cooling is required	None	1. Loss of **ALL** feedwater **AND** SG Cooling is required	None	1. Calculated Clad Temperature in **Region 3** or higher (DB-OP-02000 Figure 2) **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C **CT Radiation / RCS Activity**	1. RE 4596A or B > Table F-2 column "FC Loss" (Note 9) **OR** 2. Dose equivalent I-131 coolant activity > 300 µCi/gm	None	1. RE 4596A or B > Table F-2 column "RCS Loss" (Note 9)	None	None	1. RE 4596A or B > Table F-2 column "CT Potential Loss" (Note 9)
D **CT Integrity or Bypass**	None	None	None	None	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on Emergency Director judgment • UNISOLABLE pathway from Containment to the environment exists **OR** 2. Indications of RCS leakage outside of containment	1. Containment pressure > 50.4 psia **OR** 2. Containment hydrogen concentration > 4% **OR** 3. Containment pressure > 40 psia with < one full train, Table F-3, of containment cooling operating per design for ≥ 15 min. (Note 1)
E **ED Judgment**	1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier

Table F-2 Containment Radiation – R/hr (RE 4596A or B)			
Time After S/D (Hrs.)	**RCS Loss**	**FC Loss**	**CT Potential Loss**
0-1	1.50E+01	3.03E+03	1.40E+04
1-2	1.50E+01	2.56 E+03	1.18 E+04
2-8	1.50E+01	1.61 E+03	7.46 E+03
8-16	1.50E+01	1.14 E+03	5.28 E+03
16-24	1.50E+01	8.66 E+02	4.00 E+03
>24	1.50E+01	3.94 E+02	1.82 E+03

Table F-3 Containment Cooling Full Train	
Spray	**Coolers**
2	0
1	1
0	2

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. Calculated Clad Temperature in **Region 3** or higher (DB-OP-02000 Figure 2)

Basis:

Generic

This reading indicates temperatures within the core ~~are sufficient to~~have caused significant superheating of reactor coolant.

Plant-Specific

Indication of severe core cooling degradation is manifested by Calculated Clad Temperature in Region 3 or higher (DB-OP-02000 Figure 2). Figure 2, Incore T/C Temperature vs. RCS Pressure for ICC, provides indication of how serious core conditions are based upon combinations of RCS pressure and incore thermocouple temperatures. If the RCS P-T point is in Region 3, the cladding temperatures in the core may be 1400°F or higher. This is a very serious condition and may lead to significant amounts of H2 production; core damage may be unavoidable as this represents a very serious inadequate core cooling condition (ref. 1, 2).

WCAP-14969-A states, "Analyses performed for the WOG ERGs for indication of inadequate core cooling concluded that the temperature indicated by the core exit thermocouples, especially during transient heatup conditions, is always several hundred degrees lower than the fuel rod cladding temperatures. Thus, an indicated temperature of 1200°F can be translated to a peak cladding temperature on the order of 1400°F" (ref. 4).

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. Bases and Deviation Document for DB-OP-02000
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A
4. WCAP-14969-A, Westinghouse Owners Group Core Damage Assessment Guidance

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. Calculated Clad Temperature in **Region 2** or higher (DB-OP-02000 Figure 2)

Basis:

Generic

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

Plant-Specific

The average incore thermocouple temperature and RCS pressure is used to determine whether Calculated Clad Temperature is in Region 2. This corresponds to a loss of RCS subcooling with clad temperatures remaining below the point where damage is immediately likely (T_{clad} approximately 900° to 1100° F).

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. Bases and Deviation Document for DB-OP-02000
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Potential Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. Loss of **ALL** feedwater **AND** SG cooling is required

Basis:

Generic

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.

Plant-Specific

In combination with RCS Potential Loss B.1, meeting this threshold would result in a Site Area Emergency.

Loss of ALL feedwater cooling heat transfer capability when SG cooling is required indicates the ultimate heat sink function is under extreme challenge AND that the RCS barrier is also challenged (ref. 1).

The phrase AND “SG cooling is required” precludes the need for classification for conditions in which RCS pressure is less than SG pressure. For large LOCA events inside the Containment, the SGs are moot because heat removal through the containment heat removal systems takes place. Therefore, SG cooling should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification (ref. 1, 2).

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. Bases and Deviation Document for DB-OP-02000
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.B

Barrier: Fuel Clad

Category: C. CT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. RE 4596A or B > Table F-2 column "FC Loss" (Note 9)

Table F-2 Containment Radiation – R/hr (RE 4596A or B)			
Time After S/D (Hrs.)	**RCS Loss**	**FC Loss**	**CT Potential Loss**
0-1	1.50E+01	3.03E+03	1.40E+04
1-2	1.50E+01	2.56 E+03	1.18 E+04
2-8	1.50E+01	1.61 E+03	7.46 E+03
8-16	1.50E+01	1.14 E+03	5.28 E+03
16-24	1.50E+01	8.66 E+02	4.00 E+03
>24	1.50E+01	3.94 E+02	1.82 E+03

Note 9: During a main steam line break in containment or LOCA with temperature >170F, there is a potential to induce transient errors into the output of RE4596A and B during the peak rate of temperature change. Consult alternate indications. If the main steam line break is accompanied by core damage this error is insignificant (ref. 4, 5, 6, 7, 8).

Basis:

Generic

The radiation monitor reading corresponds to an instantaneous release of ALL reactor coolant mass into the containment, assuming that reactor coolant activity equals 300 µCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

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.

Plant-Specific

The containment high range monitors, RE 4596A & B., monitor the gamma dose rate resulting from a postulated loss of coolant accident (LOCA). RE 4596A & B are located inside containment. The detector range is approximately 1 to 1E8 R/hr (logarithmic scale) (ref. 1).

The Table F-2 values, column FC Loss represents, based on Calculation EP-EALCALC-DB-0701, the expected containment high range radiation monitor (RE 4596A & B) response based on a LOCA, for periods of 1, 2, 8, 16, 24 and 48 (>24) hours after shutdown with ~4.33% fuel failure (ref. 2).

When evaluating fission product barrier integrity values in Table F-2, time after shutdown should be confirmed with the Control Room as the time that the reactor is tripped, and reactor power is verified to be lowering on the Intermediate Range (ref. 3). If time after shutdown is less than one hour (or the reactor is still critical), the 0-1 hour after shutdown value should be chosen. This is conservative as it represents sufficient time for plant conditions to deteriorate to the point that core damage may occur, and the activity released from the RCS into the containment atmosphere to reach equilibrium mixing throughout containment.

During a main steam line break in containment or LOCA with temperature > 170ºF, there is a potential to induce transient errors (positive and negative) into the output of RE4596A & B during the peak rate of temperature change. Consult alternate indications. If the main steam line break or LOCA is accompanied by core damage this error is, however, insignificant (ref. 4).

DBNPS Basis Reference(s):

1. UFSAR Section 7.13.3.1 Containment High Radiation Monitors
2. EP-EALCALC-DB-0701 Containment Radiation Monitor Readings Following Clad Damage (FC2 and CT2 Potential Loss)
3. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
4. NRC Information Notice 97-45 Supplement 1 Environmental Qualification Deficiency for Cables and Containment Penetration Pigtails
5. Condition Report 09-53277, Containment High Range Rad Monitor Function and Current EALS
6. Condition Report 09-53278, Containment High Range Rad Monitor Function and NEI 99-01 EAL Submittal
7. Condition Report 07-31108, Potential for Thermally Induced Currents In Containment HRRM
8. Condition Report 09-55171, Containment High Range Rad Monitors Engineering Assistance Requested
9. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.A

Barrier: Fuel Clad

Category: C. CT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. Dose equivalent I-131 coolant activity > 300 µCi/gm

Basis:

Generic

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.

Plant-Specific

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The threshold dose equivalent I-131 concentration is well above that expected for iodine spikes and corresponds to about 4.33% fuel clad damage. When reactor coolant activity reaches this level the Fuel Clad barrier is considered lost (ref. 1).

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

DBNPS Basis Reference(s):

1. EP-EALCALC-DB-0701 Containment Radiation Monitor Readings Following Clad Damage (FC2 and CT2 Potential Loss)
2. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.B

Barrier: Fuel Clad

Category: C. CT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier

Basis:

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad Barrier is lost.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours 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 Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

DBNPS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Fuel Clad Loss 6.A

Barrier: Fuel Clad

Category: E. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier

Basis:

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours 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 Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

DBNPS 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 (SFAS) actuation required by **EITHER:**
   - UNISOLABLE RCS leakage
   - SG tube RUPTURE

Basis:

Generic

This threshold is based on an UNISOLABLE RCS leak of sufficient size to require an automatic or manual actuation of the Emergency Core Cooling System (ECCS). This condition clearly represents a loss of the RCS Barrier.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED. If a RUPTURED steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold ~~1.~~A.1 will also be met.

Plant-Specific

ECCS (SFAS) actuation is caused by (ref. 1):

- Low RCS pressure
- High Containment pressure

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. 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 Makeup Pump is required (> 250 gpm) by **EITHER:**
   - UNISOLABLE RCS leakage
   - SG tube leakage

Basis:

Generic

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.

Plant-Specific

The Makeup and Purification System includes two centrifugal makeup pumps, which take suction from the Makeup Tank, and return cooled, purified reactor coolant to the RCS. One of the two makeup pumps handles normal charging flow. Makeup pump capacity of a single makeup injection line is ~250 gpm. A second makeup pump being required is indicative of a substantial RCS leak (ref. 1, 2).

DBNPS Basis Reference(s):

1. UFSAR Table 9.3-8 Makeup and Purification System Component Data
2. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
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. PTS requirements invoked (SR5)

Basis:

Generic

This condition indicates an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized).

Plant-Specific

With an extended overcooling, thermal shock becomes a concern. Pressurized Thermal Shock (PTS) limits must be invoked if the criteria specified in Specific Rule 5 are met. The RCS pressure must be controlled to ensure that PTS limits are not violated. This requires action on the part of the operator to control RCS pressure and temperature (ref. 1).

The "Potential Loss" threshold is defined by the PTS limits as specified in DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture Specific Rule 5 being invoked. This indicates an extreme challenge to the RCS barrier (ref. 1, 2).

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. Bases and Deviation Document for DB-OP-02000
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. Loss of **ALL** feedwater

   AND

   SG cooling is required

Basis:

Generic

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.

Plant-Specific

In combination with FC Potential Loss B.2, meeting this threshold results in a Site Area Emergency.

Loss of ALL feedwater cooling heat transfer capability when SG cooling is required indicates the ultimate heat sink function is under extreme challenge and that the RCS barrier is also challenged (ref. 1).

The phrase AND ”SG cooling required” precludes the need for classification for conditions in which RCS pressure is less than SG pressure. For large LOCA events inside the Containment, the SGs are moot because heat removal through the containment heat removal systems takes place. Therefore, SG cooling should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification (ref. 1, 2).

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. Bases and Deviation Document for DB-OP-02000
3. NEI 99-01 Inadequate Heat Removal RCS Loss 2.B

Barrier: Reactor Coolant System

Category: C. CT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

1. RE 4596A or B > Table F-2 column "RCS Loss" (Note 9)

Table F-2 Containment Radiation – R/hr (RE 4596A or B)			
Time After S/D (Hrs.)	**RCS Loss**	**FC Loss**	**CT Potential Loss**
0-1	1.50E+01	3.03E+03	1.40E+04
1-2	1.50E+01	2.56 E+03	1.18 E+04
2-8	1.50E+01	1.61 E+03	7.46 E+03
8-16	1.50E+01	1.14 E+03	5.28 E+03
16-24	1.50E+01	8.66 E+02	4.00 E+03
>24	1.50E+01	3.94 E+02	1.82 E+03

Note 9: During a main steam line break in containment or LOCA with temperature >170F, there is a potential to induce transient errors into the output of RE4596A and B during the peak rate of temperature change. Consult alternate indications. If the main steam line break is accompanied by core damage this error is insignificant (ref. 5, 6, 7, 8, 9).

Basis:

Generic

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.

Plant-Specific

The containment high range monitors, RE 4596A & B., monitor the gamma dose rate resulting from a postulated loss of coolant accident (LOCA). RE 4596A & B are located inside containment. The detector range is approximately 1 to 1E8 R/hr (logarithmic scale) (ref. 1).

The Table F-2 values, column RCS Loss represents, based on Calculation EP-EALCALC-DB-0702, the expected containment high range radiation monitor (RE 4596A & B) response based on a LOCA, using the USAR maximum RCS activity (no core damage) (ref. 2, 3).

During a main steam line break in containment or LOCA with temperature > 170ºF, there is a potential to induce transient errors (positive and negative) into the output of RE4596A & B during the peak rate of temperature change. Consult alternate indications. If the main steam line break or LOCA is accompanied by core damage this error is, however, insignificant (ref. 5).

Since fuel cladding degradation and/or failures could also result in high Containment Area Radiation levels, a reading of >15 R/hr may be obtained without a physical loss of the RCS barrier. However, this threshold should be declared as being met if the Containment Area Radiation levels are >15 R/hr even if there are no other indications of a RCS leak or physical loss. In this case it would also be prudent to evaluate the fuel clad FISSION PRODUCT BARRIER THRESHOLDS.

DBNPS Basis Reference(s):

1. UFSAR Section 7.13.3.1 Containment High Radiation Monitors
2. EP-EALCALC-DB-0702 Containment Radiation Monitor Readings Following a LOCA (RC2 Loss)
3. USAR Table 15A-4 Maximum Fission Product Activity in Reactor Coolant
4. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
5. NRC Information Notice 97-45 Supplement 1 Environmental Qualification Deficiency for Cables and Containment Penetration Pigtails
6. Condition Report 09-53277, Containment High Range Rad Monitor Function and Current EALS
7. Condition Report 09-53278, Containment High Range Rad Monitor Function and NEI 99-01 EAL Submittal
8. Condition Report 07-31108, Potential for Thermally Induced Currents In Containment HRRM
9. Condition Report 09-55171, Containment High Range Rad Monitors Engineering Assistance Requested
10. NEI 99-01 CMT Radiation / RCS Activity RCS Loss 3.A

Barrier: Reactor Coolant System

Category: B. CT Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier

Basis:

Generic

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is lost.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS Barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current SAFETY SYSTEM performance. The term "IMMINENT" refers to the 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 Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

DBNPS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Loss 6.A

Barrier: Reactor Coolant System

Category: E. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier

Basis:

Generic

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS Barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current SAFETY SYSTEM performance. The term "IMMINENT" refers to the inability to reach final safety acceptance criteria before completing 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 Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

DBNPS 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

Basis:

Generic

This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for RCS Barrier Potential Loss ~~1.~~A.1 and Loss ~~1.~~A.1, respectively. This condition represents a bypass of the containment barrier.

FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably (~~[~~part of the FAULTED definition)~~]~~ and the FAULTED steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition; the steam generator is still considered FAULTED for emergency classification purposes.

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC ~~SU3~~ SU4 for the fuel clad barrier (i.e., RCS activity values) and IC ~~SU4~~ SU5 for the RCS barrier (i.e., RCS leak rate values).

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, glad seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category ~~A~~ R ICs.

The ~~emergency classification level~~ECLs resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

	Affected SG is FAULTED Outside of Containment?	
P-to-S Leak Rate	**Yes**	**No**
Less than or equal to ~~25~~ 10 gpm	No classification	No classification
Greater than ~~25~~ 10 gpm for 15 minutes or longer	Unusual Event per ~~SU4~~SU5	Unusual Event per ~~SU4~~SU5
Requires operation of a standby charging (makeup) pump (*RCS Barrier Potential Loss*)	Site Area Emergency per FS1	Alert per FA1
Requires an automatic or manual ECCS (SI) actuation (*RCS Barrier Loss*)	Site Area Emergency per FS1	Alert per FA1

There is no Potential Loss threshold associated with RCS or SG Tube Leakage.

Plant-Specific

None

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. 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. Calculated Clad Temperature in **Region 3** or higher (DB-OP-02000 Figure 2)

   AND

   Restoration procedures **not** effective within 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Basis:

Generic

This condition represents an IMMINENT core melt sequence, which, if not corrected, could lead to vessel failure and an increased potential for containment failure. For this condition to occur there must already have been a loss of the RCS Barrier AND the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.

The restoration procedure is considered “effective” if core exit thermocouple readings are decreasing and/or if reactor vessel level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The Emergency Director should escalate the emergency classification level as soon as it is determined that the procedure(s) will not be effective.

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.

Plant-Specific

Indication of severe core cooling degradation is manifested by Calculated Clad Temperature in Region 3 (DB-OP-02000 Figure 2). Figure 2, Incore T/C Temperature vs. RCS Pressure for ICC, provides indication of how serious core conditions are based upon combinations of RCS pressure AND incore thermocouple temperatures. If the RCS P-T point is in Region 3, the cladding temperatures in the core may be 1400°F or higher. This is a very serious condition

and may lead to significant amounts of H2 production; core damage may be unavoidable as this represents a very serious inadequate core cooling condition (ref. 1, 2).

The function restoration procedures are those emergency operating procedures that address the recovery of core cooling functions. The procedure is considered effective if the clad temperature is decreasing or if RCS water level is increasing (ref. 1, 2).

DBNPS Basis Reference(s):

1. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
2. Bases and Deviation Document for DB-OP-02000
3. NEI 99-01 Inadequate Heat Removal Containment Potential Loss 2.A

Barrier: Containment

Category: C. CT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: C. CT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

1. RE 4596A or B > Table F-2 column "CT Potential Loss" (Note 9)

Table F-2 Containment Radiation – R/hr (RE 4596A or B)			
Time After S/D (Hrs.)	**RCS Loss**	**FC Loss**	**CT Potential Loss**
0-1	1.50E+01	3.03E+03	1.40E+04
1-2	1.50E+01	2.56 E+03	1.18 E+04
2-8	1.50E+01	1.61 E+03	7.46 E+03
8-16	1.50E+01	1.14 E+03	5.28 E+03
16-24	1.50E+01	8.66 E+02	4.00 E+03
>24	1.50E+01	3.94 E+02	1.82 E+03

Note 9: During a main steam line break in containment or LOCA with temperature >170F, there is a potential to induce transient errors into the output of RE4596A and B during the peak rate of temperature change. Consult alternate indications. If the main steam line break is accompanied by core damage, this error is insignificant.

Basis:

Generic

The radiation monitor reading corresponds to an instantaneous release of ALL reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment, which would then escalate the ~~emergency classification level~~ECL to a General Emergency.

Plant-Specific

The containment high range monitors, RE 4596A & B., monitor the gamma dose rate resulting from a postulated loss of coolant accident (LOCA). RE 4596A & B are located inside containment. The detector range is approximately 1 to 1E8 R/hr (logarithmic scale) (ref. 1).

The Table F-2 values, column CT Potential Loss represents, based on Calculation EP-EALCALC-DB-0701, the expected containment high range radiation monitor (RE 4596A & B) response based on a LOCA, for periods of 1, 2, 8, 16, 24 and 48 (>24) hours after shutdown with ~20% fuel failure (ref. 2).

When evaluating fission product barrier integrity values in Table F-2, time after shutdown should be confirmed with the Control Room as the time that the reactor is tripped, and reactor power is verified to be lowering on the Intermediate Range (ref. 3). If time after shutdown is less than one hour (or the reactor is still critical), the 0-1 hour after shutdown value should be chosen. This is conservative as it represents sufficient time for plant conditions to deteriorate to the point that core damage may occur, and the activity released from the RCS into the containment atmosphere to reach equilibrium mixing throughout containment.

During a main steam line break in containment or LOCA with temperature > 170ºF, there is a potential to induce transient errors (positive and negative) into the output of RE4596A & B during the peak rate of temperature change. Consult alternate indications. If the main steam line break or LOCA is accompanied by core damage this error is, however, insignificant (ref. 4).

DBNPS Basis Reference(s):

1. UFSAR Section 7.13.3.1 Containment High Radiation Monitors
2. EP-EALCALC-DB-0701 Containment Radiation Monitor Readings Following Clad Damage (FC2 and CT2 Potential Loss)
3. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
4. NRC Information Notice 97-45 Supplement 1 Environmental Qualification Deficiency for Cables and Containment Penetration Pigtails
5. NEI 99-01 CMT Radiation / RCS Activity Containment Potential Loss 3.A

Barrier: Containment

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. Containment isolation is required

   AND EITHER:

   - Containment integrity has been lost based on Emergency Director judgment
   - UNISOLABLE pathway from Containment to the environment exists

Basis:

Generic

These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both bulleted thresholds ~~4.A.1 and 4.A.2~~.

~~4.A.1~~First Threshold – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the 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.).

Refer to the middle piping run of Figure ~~9-F-4~~1. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

Another example would be a loss or potential loss of the RCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment

leakage through various penetrations or system components. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category ~~A~~ R ICs.

~~4.A.2~~Second Threshold – Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment atmosphere to the environment. As used here, the term “environment” includes the atmosphere of a room or area, outside the containment, that may, in turn, communicate with the outside-the-plant atmosphere (e.g., through discharge of a ventilation system or atmospheric leakage). Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.

Refer to the top piping run of Figure ~~9-F-4~~1. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Leakage between two interfacing liquid systems, by itself, does not meet this threshold.

Refer to the bottom piping run of Figure ~~9-F-4~~1. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump developed a leak that allowed steam/water to enter the Auxiliary Building, then second threshold ~~4.B~~ would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause the first threshold ~~4.A.1~~ to be met as well.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable containment leakage through various penetrations or system components. Minor releases may also occur if a containment isolation valve(s) fails to close but the containment atmosphere escapes to an enclosed system. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category ~~A~~ R ICs.

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold ~~1.~~A.1.

Plant-Specific

None

DBNPS Basis Reference(s):

1. NEI 99-01 CMT Integrity or Bypass Containment Loss 4.A

Barrier: Containment

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

2. Indications of RCS leakage outside of Containment

Basis:

Generic

Containment sump level, 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 level, temperature, pressure, flow and/or radiation level readings outside of the containment may indicate that the RCS mass is being lost outside of containment.

Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of RCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside of containment may not increase significantly; however, other unexpected changes in sump levels, area temperatures or pressures, flow rates, etc. should be sufficient to determine if RCS mass is being lost outside of the containment.

Refer to the middle piping run of Figure ~~9-F-4~~1. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause threshold ~~4.A~~D.1 to be met as well.

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.

Plant-Specific

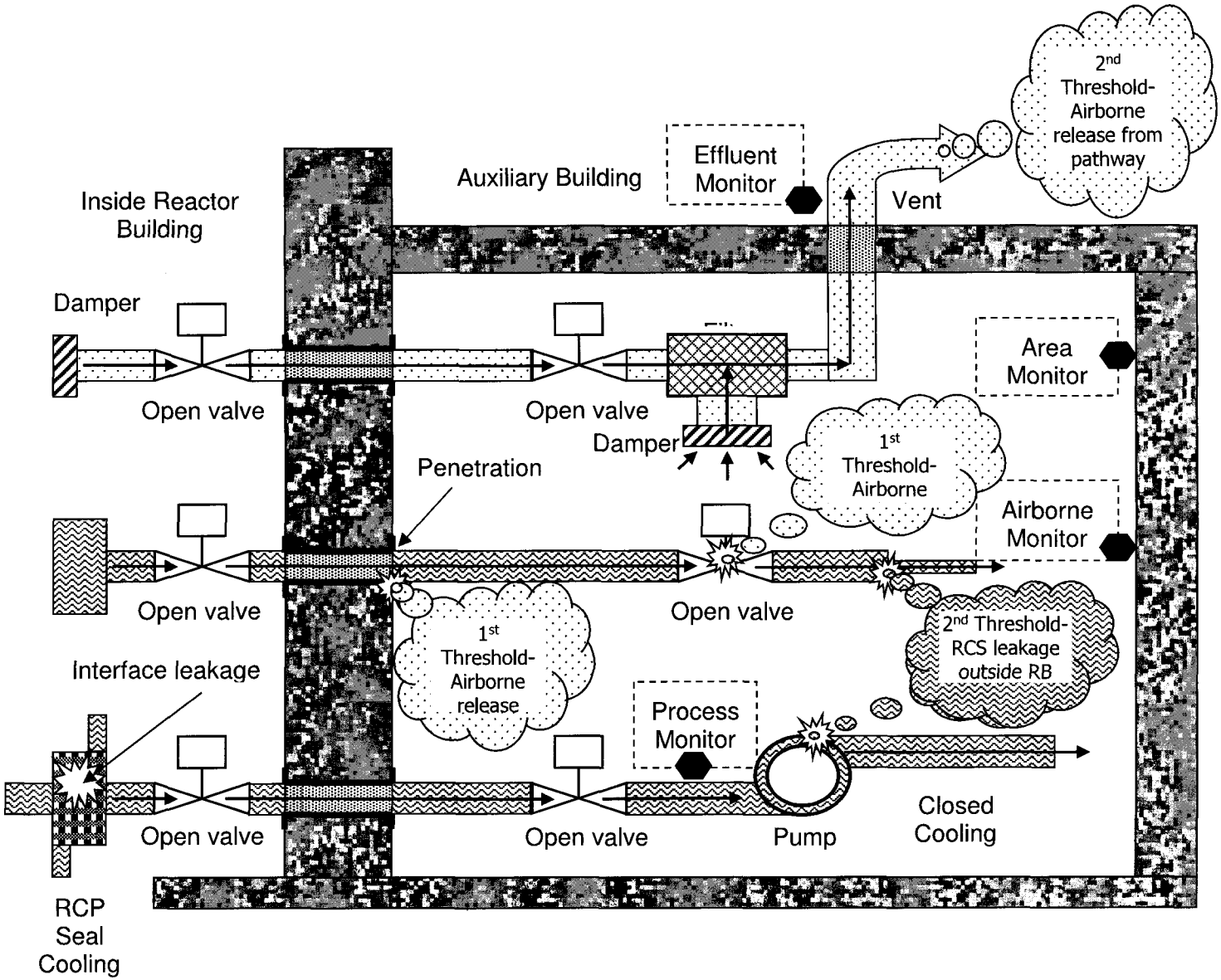
Potential RCS leak pathways outside containment include (ref. 1, 2):

- Decay Heat Removal
- ECCS (Safety Injection)
- Makeup and Purification
- RC pump seals
- RCS sample lines
- RCS drain lines

DBNPS Basis Reference(s):

1. UFSAR Section 5.2.4.7 Leakage Identification
2. DB-OP-02522 Small RCS Leaks
3. NEI 99-01 CMT Integrity or Bypass Containment Loss

Figure 1: Containment Integrity or Bypass Examples



Barrier: Containment

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

1. Containment pressure > 50.4 psia

Basis:

Generic

If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the RCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a Site Area Emergency and General Emergency since there is now a potential to lose the third barrier.

Plant-Specific

50.4 psia (36 psig + elevation adjusted atmospheric pressure of 14.4 psia) is based on the containment design pressure (ref.1).

DBNPS Basis Reference(s):

1. UFSAR Section 3.8.2.1 Containment Vessel
2. 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%

Basis:

Generic

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a potential loss of the Containment Barrier.

Plant-Specific

Following a design basis accident, hydrogen gas may be generated inside the containment by reactions such as zirconium metal with water, corrosion of materials of construction and radiolysis of aqueous solution in the core and sump (ref. 1).

The Containment Hydrogen Monitoring System is used to monitor the hydrogen concentration inside containment after a severe accident involving core damage. The containment hydrogen monitors (AI 5027 & AI 5028) are not required to be operated in the continuous mode, however, the system is required to be started up 30 minutes after containment sprays have been initiated (ref. 2).

The lower limit for the occurrence of an in-containment hydrogen burn is approximately 4% (ref. 1).

To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers must have occurred. With the Potential Loss of the containment barrier, the threshold hydrogen concentration, therefore, will likely warrant declaration of a General Emergency.

DBNPS Basis Reference(s):

1. DBSAMG-TBD Davis-Besse Severe Accident Management Guidelines Technical Bases Document
2. UFSAR Section 7.13.3.4 Containment Hydrogen Monitors
3. 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 > 40 psia with < one full train of containment cooling, Table F-3, operating per design for ≥ 15 min. (Note 1)

Table F-3 Containment Cooling Full Train	
CT Spray Pumps	**CT Cooling Fans**
2	0
1	1
0	2

Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Basis:

Generic

This threshold describes a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems (e.g., containment sprays, ice condenser fans, etc., but not including containment venting strategies) are either lost or performing in a degraded manner.

Plant-Specific

The combination of Containment spray pumps and Containment cooling fan units considered to be a full train of containment cooling operating per design is shown in Table F-3 (ref. 1).

SFAS 2 actuation automatically initiates Containment Air Coolers upon exceeding the Containment pressure high setpoint of 18.7 psia or low RCS pressure of 1600 psig. SFAS Level 4 actuation automatically initiates Containment Spray upon exceeding the Containment pressure high-high setpoint of 40 (nominal) psia (ref. 2, 3).

DBNPS Basis Reference(s):

1. UFSAR Section 6.2.2. Containment Vessel Heat Removal Systems
2. DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture
3. DBNPS Technical Specifications Table 3.3.5-1 Safety Features Actuation System Instrumentation
4. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.C

Barrier: Containment

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier

Basis:

Generic

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is lost.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours 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 Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

DBNPS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

Barrier: Containment

Category: F. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier

Basis:

Generic

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is lost.

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours 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 Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

DBNPS 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.*

DBNPS Table R-2 and H-2 Bases

NEI 99-01 Rev 06 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.

DB-OP-06902, Power Operations Rev 46 was reviewed to determine what actions are "necessary" to maintain power operations. It was determined that over reasonable periods of time (days vice years) there are no actions outside the Control Room that are required to be performed to maintain normal operations. Eventually, a shutdown would be required if Technical Specification surveillance testing was not completed and you complied with the associated LCO's 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 that an operator may be dispatched in order perform a normal plant cooldown and shutdown. The review was completed using the following procedures as the controlling documents:

DB-OP-06902, Power Operations R46

DB-OP-06903, Plant Cooldown R47

DB-OP-02504, Rapid Shutdown R20

In addition, DB-OP-06012 was reviewed to ensure the Decay Heat Removal System is aligned.

At Davis-Besse, RCS Cooldown starts once both Steam Generators reach Low Level Limits during a power reduction (approximately 30% power). As a result, this review started with DB-OP-06902, Power Operations, Section 8, Turbine and Reactor Shutdown and then transitioned to DB-OP-06903, Plant Cooldown. Each step in the controlling procedures was evaluated to determine if the action was performed in the Control Room or in the plant. In-plant actions were evaluated and a determination was made whether or not the actions, if not performed, would prevent achieving cold shutdown. The following generic assumptions were applied:

- Steps involving optional degasing of the RCS were not included since degasing the RCS is not required to reach cold shutdown.
- Steps involving supplying Auxiliary Steam were not included since AFW and AVVs can be used to reach cold shutdown if Condenser vacuum is lost.
- Steps involving Main Feedwater Pumps were not included since AFW and AVVs can be used to reach cold shutdown if Main Feedwater is not available.
- Travel paths to the locations where the equipment is operated are not part of the determination, only the rooms where the equipment is actually operated are considered

as the affected room. Travel paths were not included because most locations can be reached via alternate travel paths if required due to a localized issue.

- No assumptions made about which LPI Train is aligned for DHR Operation. Locations could be reduced by preselecting one train (typically Train 2) to provide this function. It is assumed that both trains are in a Standby LPI mode at the start of the event.

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 in EAL Tables R-2 and H-2.

The control room was not included in Table H-2 evaluation because the control room is governed by H6 series for Control Room Evacuation.

UFSAR Section 6.4.2 Toxic Gas Protection Provisions states that no toxic or explosive materials are stored in volumes or locations, which pose a control room habitability hazard that exceeds emergency system capabilities.

EAL RA3.1 addresses control room habitability relative to area radiation levels.

In Plant Task - Procedure and Step	Step Action	If action not performed, does this prevents cooldown shutdown?	Building	Elevation	Room	Mode
06902 8.2.2	Align CWRT to RCS Makeup.	No – Inventory and Boration requirements can be met from Control Room using BWST and BAATs.	N/A	N/A	N/A	N/A
06902 8.2.3	Turbine Overspeed trip testing if required.	No – This testing is not required. Testing could be performed on subsequent restart.	N/A	N/A	N/A	N/A
06902 8.3.9 Bullet	Set condenser pressure to 1.5 to 2.5 inches HgA using PCV1061.	No – This action is not required to perform Shutdown – Cooldown.	N/A	N/A	N/A	N/A
06902 8.3.9 Bullet	Attachment 8, Drain and Steam Trap Alignment for Shutdown	No – This only affects efficiency or improves moisture removal.	N/A	N/A	N/A	N/A
06902 8.3.9 Bullet	Fill and Vent the SG Blowdown lines,	No – This action facilitates SG Fill, Soak, and Drains, but is not required to complete Shutdown – Cooldown.	N/A	N/A	N/A	N/A
06902 8.3.9 Bullet	Place both Instrument Air dryers in service in parallel,	No - Step improves response of air system, but is not required.	N/A	N/A	N/A	N/A
06902 8.3.9 Bullet	MDFP is in Standby in the Main Feedwater mode and warm up is complete.	No – MDFP could be used in AFW Mode to reach Cold Shutdown.	N/A	N/A	N/A	N/A
06902 8.3.9 Bullet	Place the Auxiliary Boiler in service.	No – Aux Steam will continue to be supplied from Main Steam. If vacuum and therefore condenser is lost, steam can be dumped via AVV's.	N/A	N/A	N/A	N/A
6902 8.3.9 Bullet	Transfer Auxiliary Steam Loads from the Main Steam Reducing Station to the Auxiliary Boiler.	No – Aux Steam will continue to be supplied from Main Steam. If vacuum and therefore condenser is lost, steam can be dumped via AVV's.	N/A	N/A	N/A	N/A

In Plant Task - Procedure and Step	Step Action	If action not performed, does this prevents cooldown shutdown?	Building	Elevation	Room	Mode
6902 8.3.10 Bullet	Perform Attachment 10, MDFP Operation.	No – MDFP could be used in AFW Mode to reach Cold Shutdown.	N/A	N/A	N/A	N/A
6902 8.3.10 Bullet	Closed the actuator cylinder equalizing valve for MS4531/MS4532	No – This action facilitates SG Fill, Soak, and Drains, but is not required to complete Shutdown – Cooldown.	N/A	N/A	N/A	N/A
6902 8.3.10 Bullet	Place the SG Blowdown Lines in service.	No – This action facilitates SG Fill, Soak, and Drains, but is not required to complete Shutdown – Cooldown.	N/A	N/A	N/A	N/A
6902 8.3.16	IF Main Turbine Overspeed Testing is required,	No – This testing is not required. Testing could be performed on subsequent restart.	N/A	N/A	N/A	N/A
6902 8.3.17	Perform Attachment 13, Turbine Shutdown.	No – This action completes Turbine Shutdown, but is not required to meet Cold Shutdown.	N/A	N/A	N/A	N/A
6902 8.3.30	Open the air isolation valve that was closed in Attachment 10, MDFP Operation:	No – Since performance of Attachment 10 is not required, neither is restoration.	N/A	N/A	N/A	N/A
6902 8.3.31	Complete the shutdown of the running MFPT to Turning Gear operation.	No – desired to complete MFPT shutdown, but will not prevent reaching cold shutdown.	N/A	N/A	N/A	N/A
6902 8.3.32	Perform Attachment 8, Drain and Steam Trap Alignment for Shutdown, step 3.0.	No – This only affects efficiency or improves moisture removal.	N/A	N/A	N/A	N/A
6902 8.3.38	Transition to DB-OP-06903, Plant Cooldown	No – This is a procedure routing step.	N/A	N/A	N/A	N/A
6903 3.3.10	Perform the following to read Reactor Vessel Head O-Ring pressure.	No – Action not required to reach cold shutdown.	N/A	N/A	N/A	N/A

In Plant Task - Procedure and Step	Step Action	If action not performed, does this prevents cooldown shutdown?	Building	Elevation	Room	Mode
6903 3.12.1	Verify the requirements pertaining to containment closure control and protected equipment are initiated.	No – Action not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.12.2	Initiate Attachment 13, Preparation of MU Filter 1 for Hydrogen Peroxide Addition.	No – Action not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.13	Direct I & C Department to perform Attachment 8, Radiation Monitor Preparations for Plant Shutdown	No – Action not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.16	IF the RCS will be opened to atmosphere, THEN begin to reduce the RCS H2 concentration to less than 15 cc/kg.	No – Opening the RCS is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.17	IF the Circulating Water System is to be drained, THEN notify Chemistry to chemically shock it prior to bypassing the Cooling Tower.	No – Draining Circ Water is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.18	Remove TPCW pumps and heat exchangers from service, REFER TO DB-OP-06263, Turbine Plant Cooling Water	No – Controlling temperature and Shutdown of TPCW is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.21	Verify one of the Clean Waste Receiver Tanks is aligned…	No – The BWST and the BAATs can be used from the Control Room for RCS inventory and Boration.	N/A	N/A	N/A	N/A
6903 3.25	Begin making preparations to Start-up Containment Vessel Purge.	No – Placing CTMT Purge in service is not required to reach cold shutdown.	N/A	N/A	N/A	N/A

In Plant Task - Procedure and Step	Step Action	If action not performed, does this prevents cooldown shutdown?	Building	Elevation	Room	Mode
6903 3.25.2.a/d.3	Place/remove a second seal return cooler in/from service.	No – Controlling a second seal return cooler not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.35.2	Begin performing Attachment 5 SG Fill, Soak, and Drain.	No – This action facilitates SG Fill, Soak, and Drains, but is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.47	Shutdown one RCP.	No – The controlling procedure provides direction when CTMT is not accessible. That direction would be used.	N/A	N/A	N/A	N/A
6903 3.53	Isolate Core Flood Tank 1 perform the following:	Yes – Action is required to reduce RCS Pressure. CFT 1 Outlet Valve power restored at E11B to allow closure.	Aux	585	Rm 304 – Corridor Outside #3 MPR	1, 2, 3
6903 3.53	Isolate Core Flood Tank 2 perform the following:	Yes – Action is required to reduce RCS Pressure. CFT 2 Outlet Valve power restored at F11A to allow closure.	Aux	603	Rm 427 - #2 Electrical Penetration Room	1, 2, 3
06903 3.63	WHEN RCS pressure is between 450 psig and 425 psig THEN establish a 0/2 or 2/0 RCP combination.	No – The controlling procedure provides direction when CTMT is not accessible. That direction would be used.	N/A	N/A	N/A	N/A
6903 3.64	Control Letdown Flow - Open or throttle MU 83	No – This action aids in RCS cleanup, which is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.64.1	IF its necessary to reduce the total makeup flow to the RCS by throttling MU 58A, NORMAL MAKEUP FLOW FE MU58 SOURCE/NEEDLE,	No – This action allow lower Makeup Flow, which would only be required in a very slow cooldown. This action is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 3.66	WHEN Feedwater flow reduces to the point where level control using the Feedwater Startup Valves is difficult, Throttle FW 139 and/or FW44.	No – This action would reduce FW flow via the MFW header, which would only be required in a very slow cooldown. This action is not required to reach cold shutdown.	N/A	N/A	N/A	N/A

In Plant Task - Procedure and Step	Step Action	If action not performed, does this prevents cooldown shutdown?	Building	Elevation	Room	Mode
6903 3.68	Start actions to reduce feedwater temperature to 200F.	No – This action will allow cooldown to proceed quicker if on Main Feedwater, but is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 4.17.1	To prevent a transfer of water from the RCS to the BWST, close DH10.	Yes – Action is required to align LPI Train 1 for DHR Operations.	Aux	565	Room 236 #2 Mechanical Penetration Room	1, 2, 3
6903 4.17.1	To prevent a transfer of water from the RCS to the BWST, close DH26.	Yes – Action is required to align LPI Train 2 for DHR Operations.	Aux	565	Room 236 #2 Mechanical Penetration Room	1, 2, 3
6903 4.17.2	Close breaker BF 1130 in F11A, for DH 11	Yes – Action is required to align either LPI Train 1 or 2 for DHR Operations.	Aux	603	Rm 427 - #2 Electrical Penetration Room	1, 2, 3
6903 4.17.2	Close breaker BE 1183 in E11B, for DH 12.	Yes – Action is required to align either LPI Train 1 or 2 for DHR Operations.	Aux	585	Rm 304 – Corridor Outside #3 MPR	1, 2, 3
6903 4.17.8.d	If DH12 does not open (which is generally does not), Install jumper.	Yes – Action is required to align either LPI Train 1 or 2 for DHR Operations.	Aux	585	Rm 304 – Corridor Outside #3 MPR	1, 2, 3
6903 4.20	Begin reducing Deaerator and Feedwater temperature to less than 120F.	No – This action will allow cooldown to proceed quicker if on Main Feedwater, but is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 4.21	WHEN RCS temperature is less than 280ºF, THEN disable HPI by racking out breakers.	No – This action will prevent low temp-high pressure conditions in RCS, but is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 4.22.3	IF MSIVs are to be stroked or pinned, THEN stroke test MS100 and MS101.	No – Action is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 4.25	Place a LPI Train in service as a Decay Heat Removal Train,	Yes – This action is performed per DB-OP-06012. However, since numerous actions are required each action in that procedure was assessed separately.	N/A	N/A	N/A	N/A

In Plant Task - Procedure and Step	Step Action	If action not performed, does this prevents cooldown shutdown?	Building	Elevation	Room	Mode
6903 4.28	Remove the Auxiliary Feedwater System from service.	No – Action is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 4.30	IF SG 1 level control becomes difficult due to excess flow through SP7B, adjust FW161/139.	No – Action is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 4.32	IF SG 2 level control becomes difficult due to excess flow through SP7A, adjust FW162/44.	No – Action is not required to reach cold shutdown.	N/A	N/A	N/A	N/A
6903 4.34	WHEN RCS temperature is less than 200ºF,	No – Cold shutdown has been reached. End of review using DB-OP-06903.	N/A	N/A	N/A	N/A
6012 3.5.5	Verify the CLOSE power fuses for AC 112, DECAY HT PUMP 1-1, are removed.	No – Just prevents inadvertent start of LPI pump 1 while transferring suction. SFAS is already blocked.	N/A	N/A	N/A	N/A
6012 3.5.6	Verify BE 1187 (E11E), MV DH64 LPI-HPI CROSS CONN ISO VLV 1, is open.	No – Just prevents possible valve motor overload if stroked with RCS Suction Source.	N/A	N/A	N/A	N/A
6012 3.5.10	Open BE 1121 (E11A), MV 2733 DH PUMP 1 SUCT VLV FRM BWST.	No – Just prevents inadvertent transfer of BWST inventory to RCS if the valve opened. SFAS is already blocked.	N/A	N/A	N/A	N/A
6012 3.5.13	Close DH10*, DH PUMP 1 MINIMUM COOLDOWN ISOLATION.	Yes – This action is required to align LPI Train 1 or 2 for DHR Operations.	Aux	565	Room 236 #2 Mechanical Penetration Room	1, 2, 3
6012 3.5.17	Open BE 1126 (E11D), MV 1517 DH NORM SUCT LINE 1 ISO VLV.	No - Just prevent inadvertent loss of DHR Train 1 suction from RCS is valve is stroked closed.	N/A	N/A	N/A	N/A
6012 3.6.5	Verify the CLOSE power fuses for AD 112, DECAY HT PUMP 1-2, are removed.	No – Just prevents inadvertent start of LPI pump 2 while transferring suction. SFAS is already blocked.	N/A	N/A	N/A	N/A

In Plant Task - Procedure and Step	Step Action	If action not performed, does this prevents cooldown shutdown?	Building	Elevation	Room	Mode
6012 3.6.6	Verify BE 1195 (F11E), MV DH63 LPI-HPI CROSS CONN ISO VLV 2, is open.	No – Just prevents possible valve motor overload if stroked with RCS Suction Source.	N/A	N/A	N/A	N/A
6012 3.6.11	Open BF 1134 (F11C), MV 2734 DH PMP 2 SUCT VLV FRM BWST.	No – Just prevents inadvertent transfer of BWST inventory to RCS if the valve opened. SFAS is already blocked.	N/A	N/A	N/A	N/A
6012 3.6.14	Close DH10*, DH PUMP 1 MINIMUM COOLDOWN ISOLATION.	Yes – Action is required to align LPI Train 1 or 2 for DHR Operations.	Aux	565	Room 236 #2 Mechanical Penetration Room	1, 2, 3
6012 3.6.19	Open BF1129 (F11C), MV 1518 DH NORM SUCT LINE 2 ISO VLV.	No - Prevents inadvertent loss of DHR Train 2 suction from RCS is valve is stroked closed.	N/A	N/A	N/A	N/A
6012 3.6.27	Verify the CLOSE power fuses for AD 112.	No – If not initially removed, then reinstalling will not be required.	N/A	N/A	N/A	N/A
6012 3.7.8.a	Station an operator at DH Pump 1.	No – Operator only stationed for monitoring function. Action is not required to reach Cold Shutdown.	N/A	N/A	N/A	N/A
6012 3.7.8.d	Verify DH59, DH PUMP 1 DISCHARGE SAMPLE ISOL, is open.	No – Opening this valve provides the capability to sample RCS inventory from the DHR system.	N/A	N/A	N/A	N/A
6012 3.8.8.a	Station an operator at DH Pump 2.	No – Operator only stationed for monitoring function. Action is not required to reach Cold Shutdown.	N/A	N/A	N/A	N/A
6012 3.8.8.d	Verify DH60, DH PUMP 2 DISCHARGE SAMPLE ISOL, is open.	No – Opening this valve provides the capability to sample RCS inventory from the DHR system.	N/A	N/A	N/A	N/A

Note: The information in the above table is included for historical reference information only and based upon the procedure revision numbers referenced in the DBNPS Table R-2 and H-2 Bases summary.
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Table R-2 & H-2 Results

Table R-2 & H-2 Safe Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Aux Bldg. 565' ele. Room 236 #2 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 585' ele. Room 304 corridor outside #3 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 603' ele. Room 427 - #2 Electrical Penetration Room	1, 2, 3

Evaluation of Proposed License Amendment

Attachment 3

Davis-Besse Nuclear Power Station NEI 99-01, Revision 6, EAL Comparison Matrix

(118 Pages Follow)

Davis-Besse Nuclear Power Station
NEI 99-01 Revision 6
EAL Comparison Matrix

Revision 0

Table of Contents

Introduction

This document provides a line-by-line comparison of the Initiating Conditions (ICs), Mode Applicability and Emergency Action Levels (EALs) in NEI 99-01, Revision 6, “Development of Emergency Action Levels for Non-Passive Reactors” (ADAMS Accession Number ML110240324), and the Davis-Besse Nuclear Power Station (DBNPS) ICs, Mode Applicability and EALs. This document provides a means of assessing DBNPS differences and deviations from the NRC endorsed guidance given in NEI 99-01. Discussion of DBNPS 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. **As shown in Table 3, DBNPS took no deviations from the generic NEI 99-01 Revision 6 guidance.**

Comparison Matrix Format

The ICs and EALs discussed in this document are grouped according to NEI 99-01 Recognition Categories. Within each Recognition Category, the ICs and EALs are listed in tabular format according to the order in which they are given in NEI 99-01. Generally, each row of the comparison matrix provides the following information:

- NEI EAL/IC identifier
- NEI EAL/IC wording
- DBNPS EAL/IC identifier
- DBNPS EAL/IC wording
- Description of any differences or deviations

EAL Emphasis Techniques

Due to the width of the table columns and table formatting constraints in this document, line breaks and indentation may differ slightly from the appearance of comparable wording in the source documents. NEI 99-01 is the source document for the NEI EALs, the DBNPS EAL Technical Bases Document for the DBNPS EALs.

The print and paragraph formatting conventions summarized below guide presentation of the DBNPS EALs in accordance with the EAL writing criteria. Space restrictions in the EAL table of this document sometimes override these 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** and as well as the terms, **ANY** and **ALL**.
- Bold font is used for certain logic terms, negative terms (**not**, **cannot**, etc.).
- Upper case print is reserved for defined terms, acronyms, system abbreviations, logic terms (and, or, etc. when not used as a conjunction), annunciator window engravings.
- Three or more items in a list are normally introduced with “**ANY** of the following...” or “**ALL** of the following...” Items of the list begin with bullets when a priority or sequence is not inferred.
- The use of **AND**/**OR** logic within the same EAL has been avoided when possible. When such logic cannot be avoided, indentation and separation of subordinate contingent phrases is employed.

Global Differences

The differences listed below generally apply throughout the set of EALs and are not repeated in the Justification sections of this document. The global differences do not decrease the effectiveness of the intent of NEI 99-01.

1. The NEI phrase “Notification of Unusual Event” has been changed to “Unusual Event” or abbreviated "UE" to reduce EAL-user reading burden.
2. NEI 99-01 IC Example EALs are implemented in separate plant EALs to improve clarity and readability. For example, NEI lists all IC HU3 Example EALs under one IC. The corresponding DBNPS EALs appear as unique EALs (e.g., HU3.1 through HU3.4).
3. Mode applicability identifiers (numbers/letter) modify the NEI 99-01 mode applicability names as follows: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown, 5 - Cold Shutdown, 6 - Refueling, D - Defueled, and All. NEI 99-01 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 reduce EAL-user reading burden, DBNPS has adopted use of Boolean symbols in place of the NEI 99-01 text modifiers within the EAL wording.

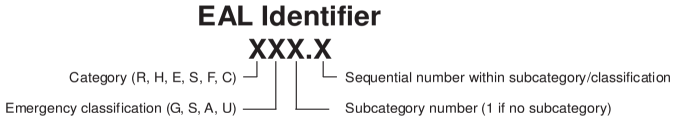
5. "min." is the standard abbreviation for "minutes" and is used to reduce EAL user reading burden.
6. The EAL Bases document has a Generic basis section and a Plant-Specific basis section. The Generic basis section provides a description of the rationale for the EAL as provided in NEI 99-01, Revision 6, and the Plant-Specific basis section provides DBNPS-relevant information concerning the EAL. The sections have been separated to reduce EAL user reading burden. The Plant-Specific and Generic basis sections were reviewed to ensure there was no conflict between the intent of the Generic NEI 99-01, Revision 6 basis and the associated Plant-Specific basis.
7. IC/EAL identification:
   - NEI Recognition Category A "Abnormal Radiation Levels/ Radiological Effluents" has been changed to Category R "Abnormal Rad Levels / Rad Effluent." 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 defines the thresholds requiring emergency classification (example EALs) and assigns them to ICs which, in turn, are grouped in "Recognition Categories."
   - The DBNPS 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, reduces EAL-user reading burden and, thereby, speeds identification of the EAL that applies to the emergency.

     b. Within each of the above three groups, assignment of EALs to categories/subcategories – Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. Subcategories are used as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The DBNPS 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



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 DBNPS 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 DBNPS 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 DBNPS proposed EAL.

Administrative changes that do not actually change the textual content are neither differences nor deviations. Likewise, any format change that does not alter the wording of the IC or EAL is considered neither a difference nor a deviation.

The following are examples of differences:

- Choosing the applicable EAL based upon plant type (i.e., BWR vs. PWR).
- Using a numbering scheme other than that provided in NEI 99-01 that does not change the intent of the overall scheme.
- Where the NEI 99-01 guidance specifically provides an option to not include an EAL if equipment for the EAL does not exist at DBNPS (e.g., automatic real-time dose assessment capability).
- Pulling information from the bases section up to the actual EAL that does not change the intent of the EAL.
- Choosing to state ALL Operating Modes are applicable instead of stating N/A, or listing each mode individually under the Abnormal Rad Level/Radiological Effluent and Hazard and Other Conditions Affecting Plant Safety sections.
- Using synonymous wording (e.g., greater than or equal to vs. at or above, less than or equal vs. at or below, greater than or less than vs. above or below, etc.)
- Adding DBNPS equipment/instrument identification and/or noun names to EALs.
- Combining like ICs that are exactly the same but have different operating modes as long as the intent of each IC is maintained and the overall progression of the EAL scheme is not affected.
- Any change to the IC and/or EAL, and/or basis wording, as stated in NEI 99-01, that does not alter the intent of the IC and/or EAL, i.e., the IC and/or EAL continues to:
    - Classify at the correct classification level.
    - Logically integrate with other EALs in the EAL scheme.

- Ensure that the resulting EAL scheme is complete (i.e., classifies all potential emergency conditions).

The following are examples of deviations:

- Use of altered mode applicability.
- Altering key words or time limits.
- Changing words of physical reference (protected area, safety-related equipment, etc.).
- Eliminating an IC. This includes the removal of an IC from the Fission Product Barrier Degradation category as this impacts the logic of Fission Product Barrier ICs.
- Changing a Fission Product Barrier from a Loss to a Potential Loss or vice-versa.
- Not using NEI 99-01 definitions as the intent is for all NEI 99-01 users to have a standard set of defined terms as defined in NEI 99-01. Differences due to plant types are permissible (BWR or PWR). Verbatim compliance to the wording in NEI 99-01 is not necessary as long as the intent of the defined word is maintained. Use of the wording provided in NEI 99-01 is encouraged since the intent is for all users to have a standard set of defined terms as defined in NEI 99-01.
- Any change to the IC and/or EAL, and/or basis wording as stated in NEI 99-01 that does alter the intent of the IC and/or EAL, i.e., the IC and/or EAL:
    - Does not classify at the classification level consistent with NEI 99-01.
    - Is not logically integrated with other EALs in the EAL scheme.
    - Results in an incomplete EAL scheme (i.e., does not classify all potential emergency conditions).

The “Difference Justification” columns in the remaining sections of this document identify each difference between the NEI 99-01 IC/EAL wording and the DBNPS IC/EAL wording. An explanation that justifies the reason for each difference is then provided. In each case, however, the differences do not decrease the effectiveness of the intent of NEI 99-01. Table 3 provides a summary list of DBNPS deviations from NEI 99-01, Revision 6, and no deviations were identified.

Table 1 – DBNPS EAL Categories/Subcategories

DBNPS EALs		NEI Recognition Category
Category	**Subcategory**	
Group: All Operating Mode:		
R – Abnormal Rad Levels/Rad Effluent	1 – Radiological Effluent 2 – Irradiated Fuel Event 3 – Area Radiation Levels	**A**bnormal Rad Levels/Radiological Effluent ICs/EALs
H – Hazards and Other Conditions Affecting Plant Safety	1 – Security 2 – Seismic Event 3 – Natural or Technological Hazard 4 – Fire 5 – Hazardous Gas 6 – Control Room Evacuation 7 – ED Judgment	**H**azards and Other Conditions Affecting Plant Safety ICs/EALs
E - Dry Fuel Storage Facility (DFSF)	1 – Confinement Boundary	ISFSI ICs/EALs
Group: Hot Conditions:		
S –System Malfunction	1 – Loss of Emergency AC Power 2 – Loss of DC Power 3 – Loss of Control Room Indications 4 – RCS Activity 5 – RCS Leakage 6 – RPS Failure 7 – Loss of Communications 8 – Containment Failure 9 – Hazardous Event Affecting Safety Systems	**S**ystem Malfunction ICs/EALs
F – Fission Product Barrier Degradation	None	**F**ission Product Barrier ICs/EALs
Group: Cold Conditions:		
C – Cold Shutdown / Refueling System Malfunction	1 – RCS Level 2 – Loss of Emergency AC Power 3 – RCS Temperature 4 – Loss of DC Power 5 – Loss of Communications 6 – Hazardous Event Affecting Safety Systems	**C**old Shutdown./ Refueling System Malfunction ICs/EALs

Table 2 – NEI / DBNPS EAL Identification Cross-Reference

NEI		DBNPS	
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		DBNPS	
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 Essential 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 Essential DC Power	CU4.1
CU5	1	C – Cold SD/ Refueling System Malfunction, 5 – Loss of Communications	CU5.1
CU5	2	C – Cold SD/ Refueling System Malfunction, 5 – Loss of Communications	CU5.2
CU5	3	C – Cold SD/ Refueling System Malfunction, 5 – Loss of Communications	CU5.3
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 Essential AC Power	CA2.1
CA3	1, 2	C – Cold SD/ Refueling System Malfunction, 3 – RCS Temperature	CA3.1

NEI		DBNPS	
IC	**Example EAL**	**Category and Subcategory**	**EAL**
CA6	1	C – Cold SD/ Refueling System Malfunction, 6 – Hazardous Event Affecting Safety Systems	CA6.1
CS1	1	N/A	N/A
CS1	2	N/A	N/A
CS1	3	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CS1.1
CG1	1	N/A	N/A
CG1	2	C – Cold SD/ Refueling System Malfunction, 1 – RCS Level	CG1.1
E-HU1	1	E – Dry Fuel Storage Facility (DFSF)	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	H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security	HU1.1
HU1	2	H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security	HU1.2
HU1	3	H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security	HU1.3
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

NEI		DBNPS	
IC	**Example EAL**	**Category and Subcategory**	**EAL**
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
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 – ED Judgment	HU7.1
HA1	1	H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security	HA1.1
HA1	2	H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security	HA1.2
HA5	1	H – Hazards and Other Conditions Affecting Plant Safety, 5 – Hazardous Gas	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 – ED 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 – ED Judgment	HS7.1
HG1	1	H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security	HG1.1
HG7	1	H – Hazards and Other Conditions Affecting Plant Safety, 7 – ED Judgment	HG7.1
SU1	1	S – System Malfunction, 1 – Loss of Essential AC Power	SU1.1
SU2	1	S – System Malfunction, 3 – Loss of Control Room Indications	SU3.1

NEI		DBNPS	
IC	Example EAL	Category and Subcategory	EAL
SU3	1	S – System Malfunction, 4 – RCS Activity	SU4.1
SU3	2	S – System Malfunction, 4 – RCS Activity	SU4.2
SU4	1	S – System Malfunction, 5 – RCS Leakage	SU5.1
SU4	2	N/A	N/A
SU4	3	N/A	N/A
SU5	1	S – System Malfunction, 6 – RPS Failure	SU6.1
SU5	2	S – System Malfunction, 6 – RPS Failure	SU6.2
SU6	1	S – System Malfunction, 7 –Loss of Communications	SU7.1
SU6	2	S – System Malfunction, 7 –Loss of Communications	SU7.2
SU6	3	S – System Malfunction, 7 –Loss of Communications	SU7.3
SU7	1	S – System Malfunction, 8 – Containment Failure	SU8.1
SU7	2	S – System Malfunction, 8 – Containment Failure	SU8.2
SA1	1	S – System Malfunction, 1 – Loss of Essential AC Power	SA1.1
SA2	1	S – System Malfunction, 3 – Loss of Control Room Indications	SA3.1
SA5	1	S – System Malfunction, 6 – RPS Failure	SA6.1
SA9	1	S – System Malfunction, 8 – Hazardous Event Affecting Safety Systems	SA9.1
SS1	1	S – System Malfunction, 1 – Loss of Essential AC Power	SS1.1
SS5	1	S – System Malfunction, 6 – RPS Failure	SS6.1

NEI		DBNPS	
IC	**Example EAL**	**Category and Subcategory**	**EAL**
SS8	1	S – System Malfunction, 2 – Loss of Essential DC Power	SS2.1
SG1	1	S – System Malfunction, 1 – Loss of Essential AC Power	SG1.1
SG8	2	S – System Malfunction, 1 – Loss of Essential AC Power	SG1.2

Table 3 – Summary of Deviations

NEI		DBNPS EAL	Description
IC	Example EAL		
None	None	N/A	N/A

Category A

Abnormal Rad Levels / Radiological Effluent

NEI IC#	NEI IC Wording and Mode Applicability	DBNPS IC#(s)	DBNPS 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 ODCM limits for 60 minutes or longer. MODE: ALL	The DBNPS ODCM is the site-specific effluent release controlling document.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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	Station Vent Channel 1 Noble Gas (RE 4598AA/BA) > 5.72E-03 µCi/cc for 60 minutes or longer **OR** **ANY** of the following effluent monitors > 2 times the high alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer: • Waste Gas System Outlet (RE 1822A or B). • Clean Waste System Outlet (RE 1770A or B). • Miscellaneous Waste System Outlet (RE 1878A or B). • Discharge permit specified monitor. (Notes 1, 2, 3)	Example EALs #1 and #2 have been combined into a single EAL to simplify presentation. EAL #1 - The NEI phrase "…effluent radiation monitor greater than 2 times the (site-specific effluent release controlling document)" has been replaced with the DBNPS site-specific effluent radiation monitor and the two times the ODCM limit. EAL #2 – The site-specific effluent radition monitors are listed.
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-	RU1.2	Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x	The DBNPS ODCM is the site-specific effluent release controlling document.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS EAL Wording	Difference Justification
	specific effluent release controlling document) limits for 60 minutes or longer.		ODCM limits for ≥ 60 min. (Notes 1, 2)	
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 Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit. Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording. The classification timeliness note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording. Deleted the phrase “due to actions to isolate the release path” and revised with “indicating that the release path is isolated” to clarify note for the reader.

NEI IC#	NEI IC Wording and Mode Applicability	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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 **ANY** of the following: • SFP level alarm of indication (LI 1600) • SFP level indication (LI 4801 A/B) • Refueling Canal low water level alarm or indication (LI 1627) • Visual observation **AND** UNPLANNED rise in corresponding area radiation levels as indicated by **ANY** of the following radiation monitors: • RE 8426 SFP Area • RE 8427 SFP Area • RE 8417 Fuel Handling Area • RE 8418 Fuel Handling Area • RE 8425 Equipment Hatch Area	The SFP and Refueling Canal low water level are the site-specific level alarms or indications. The site-specific radiation monitors are listed. The area radiation monitors are listed in bullet format for clarification.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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	Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E -02 µCi/cc for 15 minutes or longer (Notes 1, 2, 3, 4)	The DBNPS radiation monitors that detect radioactivity effluent release to the environment and their threshold is listed. DBNPS does not have a liquid threshold limit for RA1.1. The DBNPS liquid effluent limits are based on the liquid effluent concentration values given in 10CFR20 Appendix B Table 2 Column 2 (see Section 2.2.1 above). The 10CFR20 values are equivalent to the radionuclide concentrations which, if ingested continuously over the course of a year, would produce a total effective dose equivalent of 0.05 rem (50 millirem). The EPA PAGs are based on a TEDE dose from immersion, inhalation and deposition. The 10CFR20 limits and the EPA limits do not represent the same type of exposure and thus cannot be compared on a one to one basis. Additionally, significant dilution assumptions are incorporated in determining ODCM ingestion limits for liquid releases such that obtaining a dose of 10 mrem in one hour would require a discharge concentration above the effluent monitor threshold (ingestion of radioactivity from a liquid release at the site boundary is not practical).

				Thus, DBNPS EALs will not contain a liquid effluent monitor threshold value that equates to 1% of the EPA PAG.
2	Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point).	RA1.2	Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)	The site boundary is the site-specific receptor point.
3	Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point) for one hour of exposure.	RA1.3	Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure (Notes 1, 2)	The site boundary is the site-specific receptor point.
4	Field survey results indicate **EITHER** of the following at or beyond (site-specific dose receptor point): ● Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer. ● Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.	RA1.4	Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: ● Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min. ● Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation. (Notes 1, 2)	The site boundary is the site-specific receptor point. Example 2 was revised to 60 min. instead of one hour to align with previous example.
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	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

	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 2: Note 3: Note 4:	If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit. 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 pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.	The classification timeliness note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording. Deleted the phrase “due to actions to isolate the release path” and revised with “indicating that the release path is isolated” to clarify note for the reader. Incorporated site-specific EAL numbers associated with generic EAL#1.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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 as indicated by a high radiation alarm on **ANY** of the following radiation monitor indications: • RE 8426 SFP Area • RE 8427 SFP Area • RE 8417 Fuel Handling Area • RE 8418 Fuel Handling Area • RE 8425 Equipment Hatch Area • RE 8446/8447 Fuel Handling Exhaust • RE 4598AA/BA Station Vent	Deleted the NEI phrase “from the fuel” because it is redundant to the preceding phrase “irradiated fuel.” Added the phrase “by a high radiation alarm on” to clarify the indication on the site-specific radiation monitors. Site-specific list of radiation monitors are listed in bullet format for clarification. Listed in bullet format for clarification.
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.	Post-Fukushima order EA-12-051 (ref.1) required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3).

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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 for RA3.2 which is only applicable in 1 – Power Operation, 2 – Startup, 3 – Hot Standby	Revised RA3.2 mode applicability to align with mode dependent Table R-2. NEI 99-01 Revision 6 IC AA3 and HA5 prescribe the declaration of an Alert based upon IMPEDED access to rooms or areas where equipment necessary for normal plant operations, cooldown or shutdown is located. These areas are intended to be plant-operating mode dependent. Attachment 3 in the EAL Bases document provides a summary of the methodology, procedures, and locations reviewed to determine the minimum set of in-plant actions, associated locations, and operating modes necessary to shut down and cool down the reactor. The locations where those actions are performed comprise the rooms/areas in Table R-2 and H-2.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 rates > 15 mR/hr in **EITHER** of the following areas: • Control Room (RE 8430 or 8431) • CAS (RE 8435 or 8436)	No other site-specific areas requiring continuous occupancy exist at DBNPS. Revised ANY to EITHER since only 2 areas are listed RE 8430 and 8431 monitors the Control room for area radiation. The CAS area does not have installed radiation monitoring but RE 8435 and 8436 are in near proximity and can be used to approximate CAS area radiation levels.
2	An UNPLANNED event results in radiation levels that prohibit or impede access to any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode	RA3.2	An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **ANY** Table R-2 Safe Shutdown Rooms or Areas (Note 5)	The site-specific list of plant rooms or areas with entry-related mode applicability are listed in Table R-2 for clarification.

	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.	N/A	Note 5	If the equipment in the listed area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.	None

Table R-2 Safe Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Aux Bldg. 565' ele. Room 236 #2 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 585' ele. Room 304 corridor outside #3 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 603' ele. Room 427 - #2 Electrical Penetration Room	1, 2, 3

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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	Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E -01 µCi/cc for 15 minutes or longer (Notes 1, 2, 3, 4)	The DBNPS radiation monitors that detect radioactivity effluent release to the environment and their threshold are listed.
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 CDE greater than 500	RS1.3	Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: ● Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min. ● Analyses of field survey samples indicate thyroid CDE 500 mrem for 60 min. of inhalation. (Notes 1, 2)	The site boundary is the site-specific receptor point.

	mrem for one hour of inhalation.			
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 Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit. Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes. Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording. The classification timeliness note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording. Deleted the phrase “due to actions to isolate the release path” and revised with “indicating that the release path is isolated” to clarify note for the reader. Incorporated site-specific EAL numbers associated with generic EAL#1.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 1 ft.	Post-Fukushima order EA-12-051 (ref.1) required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3). Indicated level of 1 ft. on LI4801A (primary) or LI4801B (Backup) corresponds approximately to 1 ft. above the top of the SFP racks (ref. 2). This is the site-specific Level three value to account for instrument uncertainties.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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	Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E +00 µCi/cc for 15 minutes or longer (Notes 1, 2, 3, 4)	The DBNPS site specifc radiation monitors that detect radioactivity effluent release to the environment monitor and threshold values are listed.
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 > 1000 mrem TEDE or 5000 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 > 1000 mR/hr expected to continue for ≥ 60 min. ● Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation. (Notes 1, 2)	The site boundary is the site-specific receptor point.

	CDE greater than 5,000 mrem for one hour of inhalation.			
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 Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit. Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes. Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using	The classification timeliness note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording. The classification timeliness note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording. Deleted the phrase “due to actions to isolate the release path” and revised with “indicating that the release path is isolated” to clarify note for the reader. Incorporated site-specific EAL numbers associated with generic EAL#1.

			actual meteorology are available.	

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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 MODE: ALL	Top of the fuel racks is the site specific Level 3.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 1 ft. for $\geq$ 60 min. (Note 1)	Post-Fukushima order EA-12-051 (ref.1) required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3). Indicated level of 1 ft. on LI4801A (primary) or LI4801B (Backup) corresponds approximately to 1 ft. above the top of the SFP racks (ref. 2). This is the site-specific Level three value to account for instrument uncertainties.
Note	The Emergency Director should declare the General Emergency promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.		Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Category C

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS IC Wording	Difference Justification
CU1	UNPLANNED loss of (reactor vessel/RCS [*PWR*] or RCS [*BWR*]) inventory for 15 minutes or longer. MODE: Cold Shutdown, Refueling	CU1	UNPLANNED loss of RCS inventory for 15 minutes or longer MODE: 5 - Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS EAL Wording	Difference Justification
1	UNPLANNED loss of reactor coolant results in (reactor vessel/RCS [*PWR*] or RCS [*BWR*]) level less than a required lower limit for 15 minutes or longer.	CU1.1	UNPLANNED loss of reactor coolant results in RCS level less than a required lower limit for ≥ 15 min. (Note 1)	None
2	a. (Reactor vessel/RCS [*PWR*] or RCS [*BWR*]) level cannot be monitored. **AND** b. UNPLANNED increase in (site-specific sump and/or tank) levels.	CU1.2	RCS level cannot be monitored **AND** **AND EITHER** • UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT due to a loss of RCS inventory • Visual observation of UNISOLABLE RCS leakage	Site-specific applicable sumps and tanks are specified. The phrase “due to a loss of RCS inventory” has been added to the DBNPS EAL for clarification. This wording implements the intent of the NEI EAL basis which 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.” Although “Visual Observation” is neither a sump nor tank, it is included in order to implement the intent of the NEI basis, which states: “…operators may determine that an inventory loss is occurring by observing changes…”
Note	The Emergency Director should declare the Unusual Event promptly upon	N/A	Note 1: The Emergency Director should declare the event promptly upon	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

	determining that 15 minutes has been exceeded, or will likely be exceeded.		determining that time limit has been exceeded, or will likely be exceeded.	

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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 essential buses for 15 minutes or longer. MODE: 5 - Cold Shutdown, 6 - Refueling, D - Defueled	“essential” is the site-specific designation for the DBNPS emergency AC buses.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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-2, to essential 4160V buses C1 and D1 reduced to a single power source for ≥ 15 min. (Note 1) **AND** **ANY** additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS	Essential buses C1 and D1 are the site-specific emergency buses. Onsite and offsite AC power sources are tabularized in Table C-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 Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table C-2 Offsite/Onsite AC Power Sources
Offsite: • X01 • X02 • X11 (back-fed via Main Transformer) **Onsite:** • EDG1 • EDG2 • SBODG

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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 due to loss of decay heat removal capability	200°F is the site-specific Tech. Spec. cold shutdown temperature limit. Added the phrase “due to loss of decay heat removal capability” to clarify that the increase in temperature is related to such capability as specified in the generic bases.
2	Loss of **ALL** RCS temperature and (reactor vessel/RCS [*PWR*] or RCS [*BWR*]) level indication for 15 minutes or longer.	CU3.2	Loss of **ALL** RCS temperature and RCS level indication for ≥ 15 min. (Note 1)	None
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS IC Wording	Difference Justification
CU4	Loss of Vital DC power for 15 minutes or longer. MODE: Cold Shutdown, Refueling	CU4	Loss of essential DC power for 15 minutes or longer. MODE: 5 - Cold Shutdown, 6 - Refueling	Replaced the term “vital” with “essential.” The term “vital” is reserved for vital 120V AC power.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 voltage indications on Technical Specification required essential 125 VDC distribution panels for ≥ 15 min. (Note 1)	< 105 VDC is the site-specific minimum essential DC bus design 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 Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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-4 onsite communication methods	Table C-4 provides a site-specific list of onsite, ORO and NRC communications methods.
2	Loss of **ALL** of the following ORO communications methods: (site specific list of communications methods)	CU5.2	Loss of **ALL** Table C-4 ORO communication methods	
3	Loss of **ALL** of the following NRC communications methods: (site specific list of communications methods)	CU5.3	Loss of **ALL** Table C-4 NRC communication methods	

Table C-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Public Address (Gaitronics)	X		
Onsite Radios	X		
Plant Telephones	X	X	X
Commercial Telephones	X	X	X
4-Way Ringdown Circuit		X	
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System (ETS)			X

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS IC Wording	Difference Justification
CA1	Loss of (reactor vessel/RCS [*PWR*] or RCS [*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	DBNPS EAL #	DBNPS EAL Wording	Difference Justification
1	Loss of (reactor vessel/RCS [*PWR*] or RCS [*BWR*]) inventory as indicated by level less than (site-specific level).	CA1.1	Loss of RCS inventory as indicated by RCS level ≤ 0.4 ft (LI 10596)	RCS level cannot be measured below the 571 feet elevation (0.4 feet on LI 10596) which is centerline of the hot leg inlet. Should RCS level drop below this point it is assumed water level cannot be monitored other than visually.
2	a. (Reactor vessel/RCS [*PWR*] or RCS [*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 RCS [*BWR*]) inventory.	CA1.2	RCS level cannot be monitored for ≥ 15 min. (Note 1) **AND EITHER** • UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT due to a loss of RCS inventory • Visual observation of UNISOLABLE RCS leakage	Site-specific applicable sumps and tanks are specified. Although “Visual Observation” is neither a sump nor tank, it is included in order to implement the intent of the NEI basis which states: “…operators may determine that an inventory loss is occurring by observing changes…”
Note	The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded	N/A	Note 1:The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording..

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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 essential buses for 15 minutes or longer MODE: MODE: 5 - Cold Shutdown, 6 – Refueling, D - Defueled	“essential” is the site-specific designation for the DBNPS emergency AC buses.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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-2, to essential 4160V buses C1 and D1 for ≥ 15 min. (Note 1)	Essential buses C1 and D1 are the site-specific emergency buses. Onsite and offsite AC power sources are tabularized in Table C-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 Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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-3 duration (Note 1, 10) **OR** UNPLANNED RCS pressure increase > 10 psig due to a loss of RCS cooling (This EAL does not apply during water-solid plant conditions).	Example EALs #1 and #2 have been combined in one DBNPS EAL for simplification. 200°F is the site-specific Tech. Spec. cold shutdown temperature limit. Table C-3 is the site-specific implementation of the generic RCS Reheat Duration Threshold table. 10 psig is the site-specific pressure increase readable by Control Room indications.
2	UNPLANNED RCS pressure increase greater than (site-specific pressure reading). (This EAL does not apply during water-solid plant conditions. [*PWR*])			
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

N/A	N/A	N/A	Note 10: 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.	Added note to provide guidance for classification should reliable RCS temperature be lost due to loss of decay hear removal flow.

Table: RCS Heat-up Duration Thresholds		
RCS Status	**Containment Closure Status**	**Heat-up Duration**
Intact (but not at reduced inventory [*PWR*])	Not applicable	60 minutes*
Not intact (or at reduced inventory [*PWR*])	Established	20 minutes*
	Not Established	0 minutes
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

Table C-3: RCS Heat-up Duration Thresholds		
RCS Status	**Containment Closure Status**	**Heat-up Duration**
Intact (but **not** reduced inventory)	N/A	60 min.*
Not intact **OR** At reduced inventory	established	20 min.*
	not established	0 min.
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable.		

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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.	CA6.1	The occurrence of **ANY** Table C-5 Hazardous Event **AND EITHER:** • Event has caused indications of degraded performance in at least one train of a SAFETY SYSTEM required for the current operating mode • The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure required for the current operating mode	The hazardous events have been listed in Table C-5 to improve the readability of the DBNPS EAL. The NEI list of hazardous events includes all DBNPS hazardous events. No additional hazardous events could be identified. Deleted the word “damage” in the first condition as that presupposes the determination of damage, but the criteria is degraded performance of a safety system train. Replaced the word “needed” with “required” consistent with technical specifications. Shift Manager has been replaced with Emergency Director as the site-specific terminology.

	OR 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.			

Table C-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Emergency Director

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS IC Wording	Difference Justification
CS1	Loss of (reactor vessel/RCS [*PWR*] or RCS [*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	DBNPS EAL #	DBNPS EAL Wording	Difference Justification
1	a. CONTAINMENT CLOSURE not established. **AND** b. (Reactor vessel/RCS [*PWR*] or RCS [*BWR*]) level less than (site-specific level).	N/A	N/A	RCS level cannot be measured below the 571 feet elevation (0.4 feet on LI 10596) which is centerline of the hot leg inlet. Should RCS level drop below this point it is assumed water level cannot be monitored other than visually.
2	a. CONTAINMENT CLOSURE established. **AND** b. (Reactor vessel/RCS [*PWR*] or RCS [*BWR*]) level less than (site-specific level).	N/A	N/A	RCS level cannot be measured below the 571 feet elevation (0.4 feet on LI 10596) which is centerline of the hot leg inlet. Should RCS level drop below this point it is assumed water level cannot be monitored other than visually.
3	a. (Reactor vessel/RCS [*PWR*] or RCS [*BWR*]) level cannot be monitored for 30 minutes or longer. **AND** b. Core uncovery is indicated by **ANY** of the following:	CS1.1	RCS level cannot be monitored for ≥ 30 min. (Note 1) **AND** Core uncovery is indicated by **ANY** of the following: • UNPLANNED increase in Containment Sumps,	Site-specific applicable sumps and tanks are specified. Site-specific radiation monitors and site-specific value are specfied. The dose rate due to core shine when the top of the core becomes uncovered should result in the specified indications on the listed monitors. No other site-specific indicatios were identified.

	• (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)		Auxiliary Building Sumps, BWST or RCDT levels of sufficient magnitude to indicate core uncovery • Containment Radiation Monitor (RE 4596A or B) reading > 16 R/hr • Refueling Bridge Portable Area Radiation Monitor reading > 30 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 Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS IC Wording	Difference Justification
CG1	Loss of (reactor vessel/RCS [*PWR*] or RCS [*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	DBNPS EAL #	DBNPS EAL Wording	Difference Justification
1	a. (Reactor vessel/RCS [*PWR*] or RCS [*BWR*]) level less than (site-specific level) for 30 minutes or longer. **AND** b. **ANY** indication from the Containment Challenge Table (see below).	N/A	N/A	RCS level cannot be measured below the 571 feet elevation (0.4 feet on LI 10596) which is centerline of the hot leg inlet. Should RCS level drop below this point it is assumed water level cannot be monitored other than visually.
2	a. (Reactor vessel/RCS [*PWR*] or RCS [*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)	CG1.1	RCS level cannot be monitored for ≥ 30 min. (Note 1) **AND** Core uncovery is indicated by **ANY** of the following: • UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT levels of sufficient magnitude to indicate core uncovery	Site-specific applicable sumps and tanks are specified. Site-specific radiation monitors and site-specific value are specfied. The dose rate due to core shine when the top of the core becomes uncovered should result in the specified indications on the listed monitors. No other site-specific indications were identified. The generic Containment Challenge table has been implemented as Table C-1. 4% hydrogen concentration in the presence of oxygen is the minimum necessary to support a hydrogen burn (4%).

	• Erratic source range monitor indication [*PWR*] • UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery • (Other site-specific indications) **AND** c. **ANY** indication from the Containment Challenge Table (see below).		• Containment Radiation Monitor (RE 4596A or B) reading > 16 R/hr • Refueling Bridge Portable Area Radiation Monitor reading > 30 R/hr • Erratic Source Range Monitor indication **AND** **ANY** Containment Challenge indication, Table C-1	
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 Emergency Director 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 and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording. Note 6 implements the asterisked note associated with the generic Containment Challenge table.

Containment Challenge Table
■ CONTAINMENT CLOSURE not established*
■ (Explosive mixture) exists inside containment
■ UNPLANNED increase in containment pressure
■ Secondary containment radiation monitor reading above (site-specific value) [*BWR*]

* If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

Table C-1 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	DBNPS IC#(s)	DBNPS 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. DBNPS is not a defueled station.

Category E

Independent Spent Fuel Storage Installation (ISFSI)

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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 canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel HSM cask > **EITHER** of the following: • 100 mrem/hr (neutron + gamma) on the HSM cask wall or roof • 100 mrem/hr (neutron + gamma) on the center of the HSM cask door	The term “cask” was replaced with “canister” The values shown represent 2 times the limits specified in the DFSF Site CSAR for radiation external to a loaded DSC.

Category F

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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 third barrier (Table F-1)	Table F-1 provides the fission product barrier loss and potential loss thresholds.

PWR Fuel Clad Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI Threshold Wording	DBNPS FPB #(s)	DBNPS FPB Wording	Difference Justification
FC Loss 1	**RCS or SG Tube Leakage** Not Applicable	N/A	None	N/A
FC Loss 2	**Inadequate Heat Removal** A.Core exit thermocouple readings greater than (site-specific temperature value).	FC Loss B.1	Calculated Clad Temperature in **Region 3** or higher (DB-OP-02000 Figure 2)	DBNPS does not use core exit thermocouples. DBNPS uses in-core thermocouples to develop calculated clad tempertures which are approximately equivalent to the generic bases. Indication of severe core cooling degradation is manifested by Calculated Clad Temperature in Region 3 (DB-OP-02000 Figure 2). Figure 2, Incore T/C Temperature vs. RCS Pressure for ICC, provides indication of how serious core conditions are based upon combinations of RCS pressure and incore thermocouple temperatures. If the RCS P-T point is in Region 3, the cladding temperatures in the core may be 1400°F or higher.
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	1. RE 4596A or B > Table F-2 column "FC Loss" (Note 9) **OR**	The Table F-2 values, column FC Loss represents, based on Calculation EP-EALCALC-DB-0701, the expected containment high range radiation monitor (RE 4596A & B) response based on a LOCA, for periods of 1, 2, 8, 16, 24 and 48 (>24) hours after shutdown with ~4.33% fuel failure. Note 9 provides guidance on the effects of high temperature on the containment radiation monitors.
		FC Loss C.2	2. Dose equivalent I-131 coolant activity > 300 µCi/gm	None
FC Loss 4	**CNMT Integrity or Bypass** Not Applicable	N/A	None	N/A

NEI FPB#	NEI Threshold Wording	DBNPS FPB #(s)	DBNPS FPB Wording	Difference Justification
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 DBNPS.
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 Emergency Director that indicates Loss of the Fuel Clad Barrier	None
FC P-Loss 1	**RCS or SG Tube Leakage** A.RCS/reactor vessel level less than (site-specific level)	FC P-Loss A.1	None	DBNPS EOPs do not utilize RCS level as a parameter indicating a challenge to core heat removal and implementation of functional restoration procedures.
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	1. Calculated Clad Temperature in **Region 2** or higher (DB-OP-02000 Figure 2) **OR**	DBNPS does not use core exit thermocouples. DBNPS uses in-core thermocouples to develop calculated clad temperatures, which are approximately equivalent to the generic bases. The average incore thermocouple temperature and RCS pressure is used to determine whether Calculated Clad Temperature is in Region 2. This corresponds to a loss of RCS subcooling with clad temperatures remaining below the point where damage is immediately likely (T_{clad} approximately 900° to 1100° F).
		FC P-Loss B.2	2. Loss of **ALL** feedwater **AND** SG cooling is required	Loss of all feedwater cooling heat transfer capability when a SG cooling is required indicates the ultimate heat sink function is under extreme challenge and that the RCS barrier is also challenged.
FC P-Loss 3	**RCS Activity/CMNT Rad** Not Applicable	N/A	None	N/A

NEI FPB#	NEI Threshold Wording	DBNPS FPB #(s)	DBNPS FPB Wording	Difference Justification
FC P-Loss 4	**CNMT Integrity or Bypass** Not Applicable	N/A	None	N/A
FC P-Loss 5	**Other Indications** A.(site-specific as applicable)	N/A	N/A	No other site-specific Fuel Clad Potential Loss indication has been identified for DBNPS.
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 Emergency Director that indicates Potential Loss of the Fuel Clad Barrier	None

Table F-2 Containment Radiation – R/hr (RE 4596A or B)			
Time After S/D (Hrs.)	**RCS Loss**	**FC Loss**	**CT Potential Loss**
0-1	1.50E+01	3.03E+03	1.40E+04
1-2	1.50E+01	2.56 E+03	1.18 E+04
2-8	1.50E+01	1.61 E+03	7.46 E+03
8-16	1.50E+01	1.14 E+03	5.28 E+03
16-24	1.50E+01	8.66 E+02	4.00 E+03
>24	1.50E+01	3.94 E+02	1.82 E+03

PWR RCS Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI IC Wording	DBNPS FPB #(s)	DBNPS 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 (SFAS) actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	SFAS is the site-specific nomenclature for SI.
RCS Loss 2	**Inadequate Heat Removal** Not Applicable	N/A	None	N/A
RCS Loss 3	**RCS Activity/CMNT Rad** A.Containment radiation monitor reading greater than (site-specific value).	RCS Loss C.1	RE 4596A or B > Table F-2 column “RCS Loss” (Note 9)	The Table F-2 values, column RCS Loss represents, based on Calculation EP-EALCALC-DB-0702, the expected containment high range radiation monitor (RE 4596A & B) response based on a LOCA, using the USAR maximum RCS activity (no core damage). Note 9 provides guidance on the effects of high temperature on the containment radiation monitors.
RCS Loss 4	**CNMT Integrity or Bypass** Not Applicable	N/A	None	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 DBNPS.

NEI FPB#	NEI IC Wording	DBNPS FPB #(s)	DBNPS 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 Emergency Director 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	1. Operation of a standby Makeup Pump (> 250 gpm) is required by **EITHER:** • UNISOLABLE RCS leakage • SG tube leakage **OR**	Normal charging flow is handled by one of the two makeup pumps. Makeup pump capacity of a single makeup injection line is ~250 gpm. A second makeup pump being required is indicative of a substantial RCS leak.
		RCS P-Loss A.2	2. PTS requirements invoked (SR5)	This threshold is defined by the PTS limits as specified in DB-OP-02000 RPS, SFAS, SFRCS Trip, or SG Tubing Rupture Specific Rule 5 being invoked. This indicates an extreme challenge to the RCS barrier.
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	Loss of **ALL** feedwater **AND** SG cooling is required	Loss of all feedwater cooling heat transfer capability when SG cooling is required indicates the ultimate heat sink function is under extreme challenge and that the RCS barrier is also challenged.
RCS P-Loss 3	**CS Activity/CMNT Rad** Not Applicable	N/A	None	N/A

NEI FPB#	NEI IC Wording	DBNPS FPB #(s)	DBNPS FPB Wording	Difference Justification
RCS P-Loss 4	**CNMT Integrity or Bypass** Not Applicable	N/A	None	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 DBNPS.
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 Emergency Director that indicates Potential Loss of the RCS Barrier	None

PWR Containment Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI IC Wording	DBNPS FPB #(s)	DBNPS FPB Wording	Difference Justification
CNMT Loss 1	**RCS or SG Tube Leakage** A.A leaking or RUPTURED SG is FAULTED outside of containment.	CNMT Loss A.1	A leaking or RUPTURED SG is FAULTED outside of containment	None
CNMT Loss 2	**Inadequate Heat Removal** Not Applicable	N/A	None	N/A
CNMT Loss 3	**RCS Activity/CMNT Rad** Not applicable	N/A	None	N/A
CNMT Loss 4	**CNMT Integrity or Bypass** A. Containment isolation is required **AND** **EITHER** of the following: 1. Containment integrity has been lost based on Emergency Director judgment. **OR** 2. UNISOLABLE pathway from the containment to the environment exists. **OR** B. Indications of RCS leakage outside of containment.	CNMT Loss D.1	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on Emergency Director judgment • UNISOLABLE pathway from Containment to the environment exists **OR**	None
		CNMT Loss D.2	2. Indications of RCS leakage outside of containment	None

NEI FPB#	NEI IC Wording	DBNPS FPB #(s)	DBNPS FPB Wording	Difference Justification
CNMT Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific Containment Loss indication has been identified for DBNPS.
CNMT Loss 6	**Emergency Director Judgment** **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	CNMT Loss E.1	**ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier	None
CNMT P-Loss 1	**RCS or SG Tube Leakage** Not Applicable	N/A	None	N/A
CNMT P-Loss 2	**Inadequate Heat Removal** A. 1. (Site-specific criteria for entry into core cooling restoration procedure) **AND** 2. Restoration procedure not effective within 15 minutes.	CNMT P-Loss B.1	Calculated Clad Temperature in **Region 3** or higher (DB-OP-02000 Figure 2) **AND** Restoration procedures **not** effective within 15 min. (Note 1)	Indication of severe core cooling degradation is manifested by Calculated Clad Temperature in Region 3 (DB-OP-02000 Figure 2). Figure 2, Incore T/C Temperature vs. RCS Pressure for ICC, provides indication of how serious core conditions are based upon combinations of RCS pressure and incore thermocouple temperatures. The function restoration procedures are those emergency operating procedures that address the recovery of core cooling functions. Added Note 1 consistent with other thresholds with a timing component.

NEI FPB#	NEI IC Wording	DBNPS FPB #(s)	DBNPS FPB Wording	Difference Justification
CNMT P-Loss 3	**RCS Activity/CMNT Rad** A. Containment radiation monitor reading greater than (site-specific value).	CNMT P-Loss C.1	RE 4596A or B > Table F-2 column "CT Potential Loss" (Note 9)	The Table F-2 values, column CT Potential Loss represents, based on Calculation EP-EALCALC-DB-0701, the expected containment high range radiation monitor (RE 4596A & B) response based on a LOCA, for periods of 1, 2, 8, 16, 24 and 48 (>24) hours after shutdown with ~20% fuel failure Note 9 provides guidance on the effects of high temperature on the containment radiation monitors.
CNMT P-Loss 4	**CNMT Integrity or Bypass** A. Containment pressure greater than (site-specific value) **OR** B. Explosive mixture exists inside containment **OR** C. 1. Containment pressure greater than (site-specific pressure setpoint) **AND** 2. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer.	CNMT P-Loss D.1	1. Containment pressure > 50.4 psia **OR**	50.4 psia (36 psig + elevation adjusted atmospheric pressure of 14. 4 psia) is based on the containment design pressure
		CNMT P-Loss D.2	2. Containment Hydrogen concentration > 4% **OR**	The lower limit for the occurrence of an in-containment hydrogen burn is approximately 4%.
		CNMT P-Loss D.3	3. Containment pressure > 40 psia with < one full train, Table F-3, of containment cooling operating per design for ≥ 15 min. (Note 1)	SFAS 2 actuation automatically initiates Containment Air Coolers upon exceeding the Containment pressure high setpoint of 18.7 psia or low RCS pressure of 1600 psig. SFAS Level 4 actuation automatically initiates Containment Spray upon exceeding the Containment pressure high-high setpoint of 40 (nominal) psia. Added Note 1 consistent with other thresholds with a timing component.
CNMT P-Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific Containment Potential Loss indication has been identified for DBNPS.

NEI FPB#	NEI IC Wording	DBNPS FPB #(s)	DBNPS FPB Wording	Difference Justification
CNMT P-Loss 6	**Emergency Director Judgment** A.**ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.	CNMT P-Loss E.1	**ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier	None

Table F-3 Containment Cooling Full Train	
Spray	**Coolers**
2	0
1	1
0	2

Category H

Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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 Security Shift Supervisor	The site-specfic security shift supervision is defined as the Security Shift Supervisor.
2	Notification of a credible security threat directed at the site.	HU1.2	Notification of a credible security threat directed at the site	
3	A validated notification from the NRC providing information of an aircraft threat.	HU1.3	A validated notification from the NRC providing information of an aircraft threat	

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS IC Wording	Difference Justification
HU2	Seismic event greater than OBE levels MODE: All	HU2	Seismic event greater than OBE levels MODE: ALL	None

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 > OBE as indicated by OBE alarm on seismic alarm panel C5764A	Immediate control room alarm indication of an earthquake of 0.08 g (OBE) or greater is annunciated on the seismic control panel (C5764A), following seismic trigger actuation by the Containment Concrete Foundation accelerograph.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS IC Wording	Difference Justification
HU3	Hazardous event. MODE: All	HU3	Hazardous event MODE: ALL	None

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS EAL Wording	Difference Justification
1	A tornado strike within the PROTECTED AREA.	HU3.1	A tornado strike within the PROTECTED AREA	None
2	Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.	HU3.2	Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode	None
3	Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).	HU3.3	Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)	None
4	A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.	HU3.4	A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)	Added reference to Note 7.
5	(Site-specific list of natural or technological hazard events)	N/A	N/A	No other site-specific hazard has been identified for DBNPS.

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.	The DBNPS note is intended to apply to generic example EAL #4, not #3 as specified in the generic guidance.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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 min. of **ANY** of the following FIRE detection indications (Note 1): ● Report from the field (i.e., visual observation) ● Receipt of multiple (more than 1) fire alarms ● Field verification of a single fire alarm **AND** The FIRE is located within **ANY** Table H-1 area	Site-specific plant rooms and areas are listed in Table H-1 to improve the readability of the EAL. Deleted “or indications” as the term is redundant to the statement above.
2	a.Receipt of a single fire alarm (i.e., no other indications of a FIRE). **AND**	HU4.2	Receipt of a single fire alarm (i.e., **no** other indications of a FIRE) **AND**	Site-specific plant rooms and areas are listed in Table H-1 to improve the readability of the EAL.

	b.The FIRE is located within **ANY** of the following plant rooms or areas: (site-specific list of plant rooms or areas) **AND** c.The existence of a FIRE is not verified within 30-minutes of alarm receipt.		The fire alarm is indicating a FIRE within **ANY** Table H-1 area **AND** The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1)	Added “fire alarm is indicating a” to clarify intent of criteria.
3	A FIRE within the plant *or ISFSI* [*for plants with an ISFSI outside the plant Protected Area*] PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.	HU4.3	A FIRE within the plant PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)	DBNPS does not have an ISFSI located outside the plant Protected Area.
4	A FIRE within the plant *or ISFSI* [*for plants with an ISFSI outside the plant Protected Area*] PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.	HU4.4	A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish	DBNPS does not have an ISFSI located outside the plant Protected Area.
Note	**Note:**The Emergency Director should declare the Unusual Event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.	N/A	Note 1:The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table H-1 Safe Shutdown Fire Areas
• Containment • Control Room • Auxiliary Building • Intake Structure • Borated Water Storage Tank

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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 exist which in the judgment of the Emergency Director warrant declaration of a UE MODE: ALL	None

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 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.	None

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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 Shift Supervisor	The site-specfic security shift supervision is defined as the Security Shift Supervisor.
2	A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.	HA1.2	A validated notification from NRC of an aircraft attack threat within 30 min. of the site	

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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: 1 – Power Operation, 2 – Startup, 3 – Hot Standby	Revised mode applicability to align with mode dependent Table H-2. NEI 99-01 Revision 6 IC AA3 and HA5 prescribe the declaration of an Alert based upon IMPEDED access to rooms or areas where equipment necessary for normal plant operations, cooldown or shutdown is located. These areas are intended to be plant-operating mode dependent. Attachment 3 in the EAL Bases document provides a summary of the methodology, procedures, and locations reviewed to determine the minimum set of in-plant actions, associated locations, and operating modes necessary to shut down and cool down the reactor. The locations where those actions are performed comprise the rooms/areas in Table R-2 and H-2.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 Safe Shutdown Rooms or Areas **AND** Entry into the room or area is prohibited or IMPEDED (Note 5)	Plant rooms or areas with entry-related mode applicability are listed in Table H-2 to improve the readability of the EAL. The control room was not included in Table H-2 because in the event of control room evacuation shutdown activities can be completed from the Auxillary Shutdown Panel. UFSAR Section 7.4.1.6 Auxiliary Shutdown Panel states that if temporary evacuation of the control room is required due to some abnormal station condition, the operator can establish and maintain the station in a safe hot shutdown condition through the use of an Auxiliary Shutdown Panel located outside of the control room.
Note	**Note:**If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then	N/A	Note 5:If the equipment in the listed area was already inoperable or out-of-service before the event	None

	no emergency classification is warranted.		occurred, then **no** emergency classification is warranted.	

Table H-2 Safe Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Aux Bldg. 565’ ele. Room 236 #2 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 585’ ele. Room 304 corridor outside #3 Mechanical Penetration Room	1, 2, 3
Aux Bldg. 603’ ele. Room 427 - #2 Electrical Penetration Room	1, 2, 3

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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 has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel	Auxiliary Shutdown Panel is the DBNPS site-specific remote shutdown panels and local control stations.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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 which in the judgment of the Emergency Director warrant declaration of an Alert MODE: ALL	None

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 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.	None

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS IC Wording	Difference Justification
HS1	HOSTILE ACTION within the PROTECTED AREA MODE: All	HS1	HOSTILE ACTION within the PROTECTED AREA MODE: ALL	None

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 PROTECTED AREA as reported by the Security Shift Supervisor	The site-specfic security shift supervision is defined as the Security Shift Supervisor.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	Revised mode applicability to align with site-specific mode restrictions, and none of the three listed safety functions are required when the reactor is defueled. In addition, per the plant USAR the DBNPS is designed to be maintained in a safe hot shutdown condition from outside the control until access to the control room is regained and the need for taking the station to a cold shutdown condition from outside the control room is not anticipated.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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*] / RCS water level [*BWR*] ● RCS heat removal	HS6.1	An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel **AND** Control of **ANY** of the following key safety functions is not reestablished within 15 min. (Note 1): ● Reactivity ● Core cooling (RCS inventory) ● RCS heat removal (ability to maintain heat sink)	Auxiliary Shutdown Panel is the DBNPS site-specific remote shutdown panels and local control stations. Added parentheticals after Core cooling and RCS heat removal for clarification. Deleted the word "control" after "reactivity" as it is redundant. 15 minutes is the DBNPS site-specific number of minutes. Added (ability to maintain heat sink) after RCS heat removal to clarify.
Note	**Note:**The Emergency Director should declare the Unusual Event promptly upon determining that the applicable time has been	N/A	Note 1:The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

	exceeded, or will likely be exceeded.			

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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 exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency MODE: ALL	None

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 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.	None

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS IC Wording	Difference Justification
HG1	HOSTILE ACTION resulting in loss of physical control of the facility. MODE: All	HG1	HOSTILE ACTION resulting in loss of physical control of the facility MODE: ALL	None

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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*]/RCS water level [*BWR*] ● RCS heat removal **OR** 2. Damage to spent fuel has occurred or is IMMINENT.	HG1.1	A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor **AND EITHER** of the following has occurred: **ANY** of the following safety functions **cannot** be controlled or maintained ● Reactivity ● Core cooling (RCS inventory) ● RCS heat removal (ability to maintain heat sink) **OR** Damage to spent fuel has occurred or is IMMINENT	The site-specfic security shift supervision is defined as the Security Shift Supervisor. Deleted the word "control" after "reactivity" as it is redundant. Added parentheticals after Core cooling and RCS heat removal for clarification. Added (ability to maintain heat sink) after RCS heat removal to clarify.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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 Emergency Director warrant declaration of a General Emergency MODE: ALL	None

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 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.	None

Category S

System Malfunction

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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 essential buses for 15 minutes or longer MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	“essential” is the site-specific designation for the DBNPS emergency AC buses.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 essential 4160V buses C1 and D1 for ≥ 15 min. (Note 1)	Essential buses C1 and D1 are the site-specific emergency buses. Table S-1 tabularizes the onsite and offsite AC power sources.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

Table S-1 Offsite/Onsite AC Power Sources
Offsite: • X11 • X11 (back-fed via Main Transformer) • X01 • X02 **Onsite:** • EDG1 • EDG2 • SBODG

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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 SAFETY SYSTEM parameters from within the Control Room for ≥ 15 min. (Note 1)	The site-specific Safety System Parameter list is tabulated 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 Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

[*BWR parameter list*]	[*PWR parameter list*]
Reactor Power	Reactor Power
RCS Water Level	RCS Level
RCS 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 pressure • In-core T/C temperature • Level in at least one S/G • Auxiliary or emergency feed flow

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS IC Wording	Difference Justification
SU3	Reactor coolant activity greater than Technical Specification allowable limits. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU4	Reactor coolant activity greater than Technical Specification allowable limits MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS EAL Wording	Difference Justification
1	(Site-specific radiation monitor) reading greater than (site-specific value).	SU4.1	Letdown Monitor (RE 1998) reading > 2.0E+06 cpm	The DBNPS historical highest failed fuel level of 0.007% corresponds to a reading of 1.4E+05 cpm on the Failed Fuel Monitor (RE 1998). The top of calibrated range for RE 1998 is 1.0E+07. A monitor value of 2.0E+06 (0.1% clad damage) was chosen for its ability to be recognized even though exceeding Technical Specification Limits could potentially result in much higher readings.
2	Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications.	SU4.2	RCS activity > Technical Specification LCO 3.4.16 as indicated by **ANY** of the following: • Dose equivalent I-131 in the unacceptable region of Figure 3.4.16-1 • > 1 µCi/gm dose equivalent I-131 for > 48 hrs. • > 100/Ē µCi/gm gross specific coolant activity	TS LCO 3.4.16 specifies the allowed RCS activity limits.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS EAL Wording	Difference Justification
1	RCS unidentified or pressure boundary leakage greater than (site-specific value) for 15 minutes or longer.	SU5.1	RCS leakage > 10 gpm for ≥ 15 min. (Note 1)	The DBNPS Technical Specification defines unidentified leakage as all RCS leakage that is not identified, regardless of where that leakage may be (either inside or outside containment). Deleted “unidentified” to align with site-specific terminology. EAL #2 was not included. DBNPS does not have the capability to classify leakage as identified leakage within 15 minutes. Therefore, for the purpose of this IC and EAL all RCS leakage is considered unidentified leakage and the 10 gpm leak rate applies. EAL #3 was not included. The DBNPS Technical Specification defines unidentified leakage as all RCS leakage that is not identified, regardless of where that leakage may be (either inside or outside containment) and therefore EAL #3 was combined with EAL #1.
2	RCS identified leakage greater than (site-specific value) for 15 minutes or longer.	N/A	N/A	
3	Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer.	N/A	N/A	
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 Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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** RPS setpoint is exceeded **AND** A subsequent manual trip action taken at the Controls Area (manual RPS trip pushbuttons or de-energizing E2 and F2) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)	Added the phrase "... as indicated by reactor power ≥ 5% to specify the criteria under which the reactor is not shutdown." Added the phrase "... after **any** RPS setpoint is exceeded" to clarify that it is a failure of the automatic trip when a valid trip signal has been exceeded." The term "reactor control consoles" was replaced with "Controls Area". The controls area is the site-specific term for reactor controls consoles. Manual RPS trip pushbuttons or de-energizing E2 and F2 are the manual control console (Controls Area) actions taken to shut down the reactor. Reactor power below 5% is the site-specific indication of a successful reactor trip. 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).
2	a.A manual trip ([PWR] / scram [BWR]) did not shutdown the reactor. **AND**	SU6.2	A manual trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **ANY** manual trip action was initiated	Added the phrase "... as indicated by reactor power ≥ 5% to specify the criteria under which the reactor is not shutdown."

	b.**EITHER** of the following: 1. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor. **OR** 2 A subsequent automatic (trip [PWR] / scram [BWR]) is successful in shutting down the reactor.		**AND** A subsequent automatic trip or manual trip action taken at the Controls Area (manual RPS trip pushbuttons or de-energizing E2 and F2) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)	Added the phrase "... after **ANY** manual trip action was initiated" to clarify that it is a failure of any manual trip when an actual manual trip signal has been inserted." Combined conditions b.1 and b.2 into a single statement to simplify the presentation. Manual RPS trip pushbuttons or de-energizing E2 and F2 are the manual control console (Controls Area) actions taken to shut down the reactor. The term "reactor control consoles" was replaced with "Controls Area". The controls area is the site-specific term for reactor controls consoles. 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 below 5% is the site-specific indication of a successful reactor trip.
Notes	**Note:**A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.	N/A	Note 8: A manual 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.	Added the word "trip" to actions to be consistent with EAL wording.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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	Table S-4 provides a site-specific list of onsite, ORO and NRC communications methods.
2	Loss of **ALL** of the following ORO communications methods: (site-specific list of communications methods)	SU7.2	Loss of **ALL** Table S-4 ORO communication methods	
3	Loss of **ALL** of the following NRC communications methods: (site-specific list of communications methods)	SU7.3	Loss of **ALL** Table S-4 NRC communication methods	

Table S-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Public Address (Gaitronics)	X		
Onsite Radios	X		
Plant Telephones	X	X	X
Commercial Telephones	X	X	X
4-Way Ringdown Circuit		X	
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telephone System			X

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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	**ANY** penetration is not closed within 15 min. of a VALID containment isolation signal (Note 1)	Simplified the EAL into a single statement for usability.
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.	SU8.2	Containment pressure > 40 psia with < one full train of containment cooling, Table S-6, operating per design for ≥ 15 min. (Note 1)	Simplified the EAL into a single statement for usability. 40 psia is the site specific containment cooling system actuation setpoint. Table S-6 specifies the site specific combinations of CT Spray Pumps and CT Cooling Fans that constitute a full train of containment cooling.
N/A	N/A	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been	Added Note 1 to be consistent with criteria applied to EALs with a timing component.

			exceeded, or will likely be exceeded	

Table S-6 Containment Cooling Full Train	
CT Spray Pumps	**CT Cooling Fans**
2	0
1	1
0	2

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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 essential buses for 15 minutes or longer. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	“essential” is the site-specific designation for the DBNPS emergency AC buses.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 essential 4160V buses C1 and D1 reduced to a single power source for ≥ 15 min. (Note 1) **AND** **ANY** additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS	Essential buses C1 and D1 are the site-specific emergency buses. Onsite and offsite AC power sources are tabularized in Table S-1.
Note	The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	DBNPS EAL #	DBNPS 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	SA3.1	An UNPLANNED event results in the inability to monitor one or more Table S-2 SAFETY SYSTEM parameters from within the Control Room for ≥ 15 min. (Note 1) **AND** **ANY** Significant Transient is or may be in progress, Table S-3	The site-specific Safety System Parameters are listed in Table S-2. Added “or may be” to take into consideration of the possible loss of indicators relied upon to determine if a transient is in progress. The significant transient list has been tabularized in Table S-3 for ease of use.

	• Thermal power oscillations greater than 10% [*BWR*]			
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

[*BWR parameter list*]	[*PWR parameter list*]
Reactor Power	Reactor Power
RCS Water Level	RCS Level
RCS 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 pressure • In-core T/C temperature • Level in at least one S/G • Auxiliary or emergency feed flow

Table S-3 Significant Transients
• Reactor trip • Runback > 25% thermal power • Electrical load rejection > 25% electrical load • Safety injection actuation

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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 controls area are not successful in shutting down the reactor MODE: 1 - Power Operation	The term “reactor control consoles” was replaced with “Controls Area”. The controls area is the site-specific term for reactor controls consoles.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 Controls Area (manual RPS trip pushbuttons and de-energizing E2 and F2) are **not** successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8)	Added the phrase "... as indicated by reactor power ≥ 5% to specify the criteria under which the reactor is not shutdown.” Manual RPS trip pushbuttons and de-energizing E2 and F2 are the manual control console (Controls Area) actions taken to shut down the reactor. The term “reactor control consoles” was replaced with “Controls Area”. The controls area is the site-specific term for reactor controls consoles. 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 below 5% is the site-specific indication of a successful reactor trip.
Notes	**Note:**A manual action is any operator action, or set of actions, which causes the control rods to be rapidly	N/A	Note 8:A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does**	Added the word “trip” to actions to be consistent with EAL wording

	inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.		**not** include manually driving in control rods or implementation of boron injection strategies.	

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	Hazardous event affecting a SAFETY SYSTEM required for the current operating mode. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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	SA9.1	The occurrence of **ANY** Table S-5 Hazardous Event **AND EITHER**: • Event has caused indications of degraded performance in at least one train of a SAFETY SYSTEM required for the current operating mode. • The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure required for the current operating mode	The hazardous events have been listed in Table S-5 to improve the readability of the DBNPS EAL. The NEI list of hazardous events includes all DBNPS hazardous events. No additional hazardous events could be identified. Deleted the word “damage” in the first condition as that presupposes the determination of damage, but the criteria is degraded performance of a safety system train. Replaced the word “needed” with “required” consistent with technical specifications. The Emergency Director is the site specific terminology for the ERO position responsible for making emergency classifications.

	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.			

Table S-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Emergency Director

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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 power and **ALL** onsite AC power to essential buses for 15 minutes or longer MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	“essential” is the site-specific designation for the DBNPS emergency AC buses.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 essential 4160V buses C1 and D1 for ≥ 15 min. (Note 1)	Essential buses C1 and D1 are the site-specific emergency buses. Added term “capability” after power for clarification. Onsite and onsite AC power sources are tabularized 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 Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS IC Wording	Difference Justification
SS5	Inability to shutdown the reactor causing a challenge to (core cooling [*PWR*] / RCS 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	DBNPS EAL #	DBNPS 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**: • Calculated Clad Temperature in **Region 3** (DB-OP-02000 Figure 2) • MFW, AFW and MU-HPI PORV Cooling are **ALL** unavailable	Added the phrase "... as indicated by reactor power ≥ 5% to specify the criteria under which the reactor is not shutdown. 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 scram. Deleted the term “manual actions” from the second condition. For generic IC SS5, all actions to shut down the reactor can be credited, including emergency boration which is not considered a “manual” trip action. Indication of continuing core cooling degradation is manifested by Calculated Clad Temperature in Region 3 (DB-OP-02000 Figure 2). An extreme challenge to heat removal is defined as a complete loss of MFW, AFW and MU-HPI PORV Cooling.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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** essential DC power for 15 minutes or longer. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	“essential” is the site-specific designation for the DBNPS vital DC buses.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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** essential DC distribution panels D1P, D1N, D2P and D2N for ≥ 15 min. (Note 1)	< 105 VDC on essential DC distribution panels D1P, D1N, D2P and D2N are the site-specific minimum Vital DC bus design. D1P, D1N, D2P and D2N are the site-specific essential DC buses.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1:The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	SG1a	Prolonged loss of **ALL** offsite and **ALL** onsite AC power to essential buses MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	"essential" is the site-specific designation for the DBNPS emergency AC buses. Combined NEI ICs SG1 and SG8 under the loss of power category for usability. FENOC considered the end-user human factors into combining SG1 and SG8 were combined into one IC and numbering the EALs SG1.1 and SG1.2. Combining the EALs will allow the AC and DC loss of power EALs to be evaluated in series making the evaluation of loss of power more effective and efficient. Although the EAL sub-category identifier does differ from the methodology, the EAL itself and the basis do not differ from the standard emergency classification and action level scheme, thus the requirement of 10 CFR 50.47(b)(4) will continue to be met.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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 essential 4160V buses C1 and D1 **AND EITHER:** • Restoration of at least one essential bus in < 4 hours is **not** likely (Note 1) • Calculated Clad Temperature in **Region 3** (DB-OP-02000 Figure 2)	Essential buses C1 and D1 are the site-specific emergency buses. Added term "capability" after power for clarification. Onsite and onsite AC power sources are tabularized in Table S-1. Four hours is the minimum time the SBODG can run fully loaded with the minimum fuel oil tank supply. Indication of continuing core cooling degradation is manifested by Calculated Clad Temperature in Region 3 (DB-OP-02000 Figure 2)
Note	The Emergency Director should declare the General Emergency promptly upon determining that	N/A	Note 1: The Emergency Director should declare the event promptly	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

	(site-specific hours) has been exceeded, or will likely be exceeded.		upon determining that time limit has been exceeded, or will likely be exceeded.	

NEI IC#	NEI IC Wording	DBNPS IC#(s)	DBNPS 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	SG1b	Loss of **ALL** essential AC and DC power sources for 15 minutes or longer. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	"essential" is the site-specific designation for the DBNPS emergency AC and DC power sources. Combined NEI ICs SG1 and SG8 under the loss of power category. FENOC considered the end-user human factors into combining SG1 and SG8 were combined into one IC and numbering the EALs SG1.1 and SG1.2. Combining the EALs will allow the AC and DC loss of power EALs to be evaluated in series making the evaluation of loss of power more effective and efficient. Although the EAL sub-category identifier does differ from the methodology, the EAL itself and the basis do not differ from the standard emergency classification and action level scheme, thus the requirement of 10 CFR 50.47(b)(4) will continue to be met.

NEI Ex. EAL #	NEI Example EAL Wording	DBNPS EAL #	DBNPS 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.	SG1.2	Loss of **ALL** offsite and **ALL** onsite AC power capability, Table S-1, to essential 4160V buses C1 and D1 for ≥ 15 min. **AND** Loss of **ALL** 125 VDC power based on battery bus voltage indications < 105 VDC on **ALL** essential DC distribution panels D1P, D1N, D2P and D2N for ≥ 15 min. (Note 1)	Essential buses C1 and D1 are the site-specific emergency buses. Added term "capability" after power for clarification. < 105 VDC is the site-specific minimum essential DC bus design voltage. D1P, D1N, D2P and D2N are the site-specific essential DC panels. Onsite and onsite AC power sources are tabularized in Table S-1.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that	N/A	Note 1: The Emergency Director should declare the event promptly	The classification timeliness and event level note has been standardized across the DBNPS EAL scheme by referencing the "time limit" specified within the EAL wording.

	15 minutes has been exceeded, or will likely be exceeded.		upon determining that time limit has been exceeded, or will likely be exceeded.	

Evaluation of Proposed License Amendment

Attachment 4

Emergency Action Level (EAL) Wallcharts

(3 Pages Follow)

		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
H Hazards	**1** Security	**HG1** HOSTILE ACTION resulting in loss of physical control of the facility **HG1.1** 1 2 3 4 5 6 DEF A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor. **AND EITHER** of the following has occurred: **ANY** of the following safety functions **cannot** be controlled or maintained • Reactivity • Core cooling (RCS inventory) • RCS heat removal (ability to maintain heat sink) **OR** Damage to spent fuel has occurred or is IMMINENT.	**HS1** HOSTILE ACTION within the PROTECTED AREA **HS1.1** 1 2 3 4 5 6 DEF A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.	**HA1** HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes **HA1.1** 1 2 3 4 5 6 DEF A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor. **HA1.2** 1 2 3 4 5 6 DEF A validated notification from NRC of an aircraft attack threat within 30 min. of the site.	**HU1** Confirmed SECURITY CONDITION or threat **HU1.1** 1 2 3 4 5 6 DEF A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervisor. **HU1.2** 1 2 3 4 5 6 DEF Notification of a credible security threat directed at the site. **HU1.3** 1 2 3 4 5 6 DEF A validated notification from the NRC providing information of an aircraft threat.
	2 Seismic Event	None	None	None	**HU2** Seismic event greater than OBE levels **HU2.1** 1 2 3 4 5 6 DEF Seismic event > OBE as indicated by OBE alarm on seismic alarm panel C5764A.
	3 Natural or Tech. Hazard		None	None	**HU3** Hazardous event **HU3.1** 1 2 3 4 5 6 DEF A tornado strike within the PROTECTED AREA. **HU3.2** 1 2 3 4 5 6 DEF Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode. **HU3.3** 1 2 3 4 5 6 DEF 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). **HU3.4** 1 2 3 4 5 6 DEF A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7).
	4 Fire		None	**Table H-1 Safe Shutdown Fire Areas** • Containment • Control Room • Auxiliary Building • Intake Structure • Borated Water Storage Tank	**HU4** FIRE potentially degrading the level of safety of the plant **HU4.1** 1 2 3 4 5 6 DEF A FIRE is **not** extinguished within 15 min. of **ANY** of the following FIRE detection indications (Note 1): • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms • Field verification of a single fire alarm **AND** The FIRE is located within **ANY** Table H-1 area. **HU4.2** 1 2 3 4 5 6 DEF Receipt of a single fire alarm (i.e., no other indications of a FIRE). **AND** The fire alarm is indicating a FIRE within **ANY** Table H-1 area. **AND** The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1). **HU4.3** 1 2 3 4 5 6 DEF A FIRE within the plant PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1). **HU4.4** 1 2 3 4 5 6 DEF A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.
	5 Hazardous Gas	None	None	**HA5** Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown **HA5.1** 1 2 3 Release of a toxic, corrosive, asphyxiant or flammable gas into **ANY** Table H-2 Safe Shutdown Rooms or Areas. **AND** Entry into the room or area is prohibited or IMPEDED. (Note 5)	**Table H-2 Safe Shutdown Rooms/Areas** Area — Mode Aux Bldg. 565' ele. Room 236 #2 Mechanical Penetration Room — 1, 2, 3 Aux Bldg. 585' ele. Room 304 corridor outside #3 Mechanical Penetration Room — 1, 2, 3 Aux Bldg. 603' ele. Room 427 - #2 Electrical Penetration Room — 1, 2, 3
	6 Control Room Evacuation	None	**HS6** Inability to control a key safety function from outside the Control Room 1 2 3 4 **HS6.1** An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel. **AND** Control of **ANY** of the following key safety functions is not reestablished within 15 min. (Note 1): • Reactivity • Core cooling (RCS inventory) • RCS heat removal (ability to maintain heat sink)	**HA6** Control Room evacuation resulting in transfer of plant control to alternate locations 1 2 3 4 5 6 DEF **HA6.1** An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel.	
	7 Emergency Director Judgment	**HG7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of General Emergency 1 2 3 4 5 6 DEF **HG7.1** 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.	**HS7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of Site Area Emergency 1 2 3 4 5 6 DEF **HS7.1** 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.	**HA7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert 1 2 3 4 5 6 DEF **HA7.1** 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.	**HU7** Other conditions exist which in the judgment of the Emergency Director warrant declaration of a UE 1 2 3 4 5 6 DEF **HU7.1** 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.
R Abnormal Rad Release / Rad Effluent	**1** Rad Effluent	**RG1** Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE 1 2 3 4 5 6 DEF **RG1.1** Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E +00 µCi/cc for 15 minutes or longer. (Notes 1, 2, 3, 4) **RG1.2** Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY. (Note 4) **RG1.3** Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min. • Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation. (Notes 1, 2)	**RS1** Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE 1 2 3 4 5 6 DEF **RS1.1** Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E -01 µCi/cc for 15 minutes or longer. (Notes 1, 2, 3, 4) **RS1.2** Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY. (Note 4) **RS1.3** Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min. • Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation. (Notes 1, 2)	**RA1** Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE 1 2 3 4 5 6 DEF **RA1.1** Station Vent Channel 1 Noble Gas (RE 4598 AB/BB) > 8.4E -02 µCi/cc for 15 minutes or longer. (Notes 1, 2, 3, 4) **RA1.2** Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY. (Note 4) **RA1.3** Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure. (Notes 1, 2) **RA1.4** Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min. • Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation. (Notes 1, 2)	**RU1** Release of gaseous or liquid radioactivity > 2 times the ODCM limits for 60 minutes or longer 1 2 3 4 5 6 DEF **RU1.1** Station Vent Channel 1 Noble Gas (RE 4598AA/BA) > 5.72E-03 µCi/cc for 60 minutes or longer. **OR** **ANY** of the following effluent monitors > 2 times the high alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer: • Waste Gas System Outlet (RE 1822A or B). • Clean Waste System Outlet (RE 1770A or B). • Miscellaneous Waste System Outlet (RE 1878A or B). • Discharge permit specified monitor. (Notes 1, 2, 3) **RU1.2** Sample analyses for a gaseous or liquid release indicates concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2)
	2 Irradiated Fuel Event	**RG2** Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer 1 2 3 4 5 6 DEF **RG2.1** Spent fuel pool level **cannot** be restored to at least 1 ft. for ≥ 60 min. (Note 1)	**RS2** Spent fuel pool level at the top of the fuel racks 1 2 3 4 5 6 DEF **RS2.1** Lowering of spent fuel pool level to 1 ft.	**RA2** Significant lowering of water level above, or damage to, irradiated fuel 1 2 3 4 5 6 DEF **RA2.1** Uncovery of irradiated fuel in the REFUELING PATHWAY **RA2.2** Damage to irradiated fuel resulting in a release of radioactivity as indicated by a high radiation alarm on **ANY** of the following radiation monitor indications: • RE 8426 SFP Area • RE 8427 SFP Area • RE 8417 Fuel Handling Area • RE 8418 Fuel Handling Area • RE 8425 Equipment Hatch Area • RE 8446/8447 Fuel Handling Exhaust • RE 4598AA/BA Station Vent **RA2.3** Lowering of spent fuel pool level to 10 ft.	**RU2** UNPLANNED loss of water level above irradiated fuel 1 2 3 4 5 6 DEF **RU2.1** UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following: • SFP level alarm or indication (LI 1600) • SFP level indication (LI 4801 A/B) • Refueling Canal low water level alarm or indication (LI 1627) • Visual observation **AND** UNPLANNED rise in corresponding area radiation levels as indicated by **ANY** of the following radiation monitors: • RE 8426 SFP Area • RE 8427 SFP Area • RE 8417 Fuel Handling Area • RE 8418 Fuel Handling Area • RE 8425 Equipment Hatch Area
	3 Area Radiation Levels	None	None	**RA3** Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown **RA3.1** 1 2 3 4 5 6 DEF Dose rate > 15 mR/hr in **EITHER** of the following areas: • Control Room (RE 8430 or 8431) • Central Alarm Station (RE 8435 or 8436) **RA3.2** 1 2 3 An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **ANY** Table R-2 Safe Shutdown Rooms or Areas. (Note 5)	**Table R-2 Safe Shutdown Rooms/Areas** Area — Mode Aux Bldg. 565' ele. Room 236 #2 Mechanical Penetration Room — 1, 2, 3 Aux Bldg. 585' ele. Room 304 corridor outside #3 Mechanical Penetration Room — 1, 2, 3 Aux Bldg. 603' ele. Room 427 - #2 Electrical Penetration Room — 1, 2, 3
E DFSF	**1** Confinement Boundary	None	None	None	**EU1** Damage to a loaded cask CONFINEMENT BOUNDARY 1 2 3 4 5 6 DEF **EU1.1** Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel HSM cask > **EITHER** of the following: • 100 mrem/hr (neutron + gamma) on the HSM cask wall or roof • 100 mrem/hr (neutron + gamma) on the center of the HSM cask door

Notes

Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then **no** emergency classification is warranted.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is **not** required.

Note 7: This EAL does **not** apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 8: A manual trip action is **ANY** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Note 9: During a main steam line break in containment or LOCA with temperature >170°F, there is a potential to induce transient errors into the output of RE4596A and B during the peak rate of temperature change. Consult alternate indications. If the main steam line break is accompanied by core damage this error is insignificant.

Note 10: 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.

Modes:

1	2	3	4	5	6	DEF
Power Operation	Startup	Hot Standby	Hot Shutdown	Cold Shutdown	Refueling	Defueled

Davis-Besse Nuclear Power Station

FirstEnergy

RA-EP-01500 Rev. 0
EAL Classification Matrix
Page 1 of 3

ALL CONDITIONS

		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
S System Malfunct.	**1** Loss of Essential AC Power	**SG1a** Prolonged loss of **ALL** offsite and **ALL** onsite AC power to essential buses (1 2 3 4) **SG1.1** Loss of **ALL** offsite and **ALL** onsite AC power capability, Table S-1, to essential 4160V buses C1 and D1 **AND EITHER:** • Restoration of at least one essential bus in < 4 hours is **not** likely (Note 1) • Calculated Clad Temperature in **Region 3** (DB-OP-02000 Figure 2) **SG1b** Loss of **ALL** AC and DC power sources for 15 minutes or longer (1 2 3 4) **SG1.2** Loss of **ALL** offsite and **ALL** onsite AC power capability, Table S-1, to essential 4160V buses C1 and D1 for ≥ 15 min. **AND** Loss of **ALL** 125 VDC power based on battery bus voltage indications < 105 VDC on **ALL** essential DC distribution panels D1P, D1N, D2P and D2N for ≥ 15 min. (Note 1)	**SS1** Loss of **ALL** offsite and **ALL** onsite AC power to essential buses for 15 minutes or longer (1 2 3 4) **SS1.1** Loss of **ALL** offsite and **ALL** onsite AC power capability, Table S-1, to essential 4160V buses C1 and D1 for ≥ 15 min. (Note 1)	**SA1** Loss of **ALL** but one AC power source to essential buses for 15 minutes or longer (1 2 3 4) **SA1.1** AC power capability, Table S-1, to essential 4160V buses C1 and D1 reduced to a single power source for ≥ 15 min. (Note 1) **AND** **ANY** additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS	**SU1** Loss of **ALL** offsite AC power capability to essential buses for 15 minutes or longer (1 2 3 4) **SU1.1** Loss of **ALL** offsite AC power capability, Table S-1, to essential 4160V buses C1 and D1 for ≥ 15 min. (Note 1) **Table S-1 Offsite/Onsite AC Power Sources** **Offsite:** • X11 • X11 (back-fed via Main Transformer) • X01 • X02 **Onsite:** • EDG1 • EDG2 • SBODG
	2 Loss of Essential DC Power		**SS2** Loss of **ALL** essential DC power for 15 minutes or longer (1 2 3 4) **SS2.1** Loss of **ALL** 125 VDC power based on battery bus voltage indications < 105 VDC on **ALL** essential DC distribution panels D1P, D1N, D2P and D2N for ≥ 15 min. (Note 1)	None	None
	3 Loss of Control Room Indications	None	**Table S-2 Safety System Parameters** • Reactor power • RCS pressure • In-core T/C temperature • Level in at least one S/G • Auxiliary or emergency feed flow	**SA3** UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress (1 2 3 4) **SA3.1** An UNPLANNED event results in the inability to monitor one or more Table S-2 SAFETY SYSTEM parameters from within the Control Room for ≥ 15 min. (Note 1) **AND** **ANY** Significant Transient is or may be in progress, Table S-3	**SU3** UNPLANNED loss of Control Room indications for 15 minutes or longer (1 2 3 4) **SU3.1** An UNPLANNED event results in the inability to monitor one or more Table S-2 SAFETY SYSTEM parameters from within the Control Room for ≥ 15 min. (Note 1)
	4 RCS Activity	None	**Table S-3 Significant Transients** • Reactor trip • Runback > 25% thermal power • Electrical load rejection > 25% electrical load • Safety injection actuation	None	**SU4** Reactor coolant activity greater than Technical Specification allowable limits (1 2 3 4) **SU4.1** Letdown Monitor (RE 1998) reading > 2.0E+06 cpm. **SU4.2** RCS activity > Technical Specification LCO 3.4.16 as indicated by **ANY** of the following: • Dose equivalent I-131 in the unacceptable region of Figure 3.4.16-1 • > 1 µCi/gm dose equivalent I-131 for > 48 hrs. • > 100/Ē µCi/gm gross specific coolant activity
	5 RCS Leakage	None	None	None	**SU5** RCS leakage for 15 minutes or longer (1 2 3 4) **SU5.1** RCS leakage > 10 gpm for ≥ 15 min. (Note 1)
	6 RPS Failure	None	**SS6** Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal (1) **SS6.1** An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5% **AND** **ALL** actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5% **AND EITHER:** • Calculated Clad Temperature in **Region 3** (DB-OP-02000 Figure 2) • MFW, AFW and MU-HPI PORV Cooling are **all** unavailable	**SA6** Automatic or manual trip fails to shut down the reactor AND subsequent manual actions taken at the Controls Area are **not** successful in shutting down the reactor (1) **SA6.1** An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5% **AND** Manual trip actions taken at the Controls Area (manual RPS trip pushbuttons and de-energizing E2 and F2) are **not** successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8).	**SU6** Automatic or manual trip fails to shut down the reactor (1) **SU6.1** An automatic trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **ANY** RPS setpoint is exceeded **AND** A subsequent manual trip action taken at the Controls Area (manual RPS trip pushbuttons or de-energizing E2 and F2) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8). **SU6.2** A manual trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **ANY** manual trip action was initiated **AND** A subsequent automatic trip or manual trip action taken at the Controls Area (manual RPS trip pushbuttons or de-energizing E2 and F2) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8).
	7 Loss of Comm.	**Notes** **Note 1:** The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. **Note 8:** A manual trip action is ANY operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies. **Note 9:** During a main steam line break in containment or LOCA with temperature >170F, there is a potential to induce transient errors into the output of RE4596A and B during the peak rate of temperature change. Consult alternate indications. If the main steam line break is accompanied by core damage this error is insignificant.	None	**Table S-4 Communications Methods** System / Onsite / ORO / NRC Public Address (Gaitronics): X Onsite Radios: X Plant Telephones: X, X, X Commercial Telephones: X, X, X 4-Way Ringdown Circuit: X Satellite Phones: X, X Cellular Phones: X, X NRC Emergency Telephone System: X	**SU7** Loss of **ALL** onsite or offsite communications capabilities (1 2 3 4) **SU7.1** Loss of **ALL** Table S-4 onsite communication methods. **SU7.2** Loss of **ALL** Table S-4 ORO communication methods. **SU7.3** Loss of **ALL** Table S-4 NRC communication methods.
	8 CMT Failure		None	**Table S-6 Containment Cooling Full Train** CT Spray Pumps / CT Cooling Fans 2 / 0 1 / 1 0 / 2	**SU8** Failure to isolate containment or loss of containment pressure control (1 2 3 4) **SU8.1** **ANY** penetration is not closed within 15 min. of a VALID containment isolation signal (Note 1) **SU8.2** Containment pressure > 40 psia with < one full train of containment cooling, Table S-6, operating per design for > 15 min. (Note 1)
	9 Hazardous Event Affecting Safety Systems	None	**Table S-5 Hazardous Events** • Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Emergency Director	**SA9** Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode (1 2 3 4) **SA9.1** The occurrence of **ANY** Table S-5 Hazardous Event **AND EITHER:** • Event has caused indications of degraded performance in at least one train of a SAFETY SYSTEM required for the current operating mode • The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure required for the current operating mode	None
F Fission Product Barrier Degradation		**FG1** (1 2 3 4) Loss of **ANY** two barriers **AND** Loss or Potential Loss of third barrier (Table F-1)	**FS1** (1 2 3 4) Loss or Potential Loss of **ANY** two barriers (Table F-1)	**FA1** (1 2 3 4) **ANY** Loss or **ANY** Potential Loss of **EITHER** Fuel Clad or RCS (Table F-1)	None

Table F-1 Fission Product Barrier Matrix

	Fuel Clad (FC) Barrier		Reactor Coolant System (RCS) Barrier		Containment (CT) 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 (SFAS) actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	1. Operation of a standby Makeup Pump (>250 gpm) is required by **EITHER:** • UNISOLABLE RCS leakage • SG tube leakage **OR** 2. PTS requirements invoked (SR5)	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B Inadequate Heat Removal	1. Calculated Clad Temperature in **Region 3** or higher (DB-OP-02000 Figure 2)	1. Calculated Clad Temperature in **Region 2** or higher (DB-OP-02000 Figure 2) **OR** 2. Loss of **ALL** feedwater **AND** SG cooling is required	None	1. Loss of **ALL** feedwater **AND** SG cooling is required	None	1. Calculated Clad Temperature in **Region 3** or higher (DB-OP-02000 Figure 2) **AND** Restoration procedures **not** effective within 15 min. (Note1)
C CT Radiation / RCS Activity	1. RE 4596A or B > Table F-2 column "FC Loss" (Note 9) **OR** 2. Dose equivalent I-131 coolant activity > 300 µCi/gm	None	1. RE 4596A or B > Table F-2 column "RCS Loss" (Note 9)	None	None	1. RE 4596A or B > Table F-2 column "CT Potential Loss" (Note 9)
D CT Integrity or Bypass	None	**Table F-2 Containment Radiation R/hr (RE 4596A or B)** Time After S/D (hrs.) / RCS Loss / FC Loss / CT Pot. Loss 0-1 / 1.50E+01 / 3.03E+03 / 1.40E+04 1-2 / 1.50E+01 / 2.56E+03 / 1.18E+04 2-8 / 1.50E+01 / 1.61E+03 / 7.46E+03 8-16 / 1.50E+01 / 1.14E+03 / 5.28E+03 16-24 / 1.50E+01 / 8.66E+02 / 4.00E+03 >24 / 1.50E+01 / 3.94E+02 / 1.82E+03	None	**Table F-3 Containment Cooling Full Train** CT Spray Pumps / CT Cooling Fans 2 / 0 1 / 1 0 / 2	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on Emergency Director judgment • UNISOLABLE pathway from Containment to the environment exists **OR** 2. Indications of RCS leakage outside of containment	1. Containment pressure > 50.4 psia **OR** 2. Containment Hydrogen concentration > 4% **OR** 3. Containment pressure > 40 psia with < one full train, Table F-3, of containment cooling operating per design for ≥ 15 min. (Note 1)
E ED Judgment	1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier

Modes: 1 Power Operation | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refueling | DEF Defueled

Davis-Besse Nuclear Power Station
FirstEnergy

RA-EP-01500 Rev. 0
EAL Classification Matrix
Page 2 of 3
HOT CONDITIONS
RCS > 200°F)

C — Cold SD/ Refueling System Malfunct.

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
1 RCS Level	**CG1** Loss of RCS inventory affecting fuel clad integrity with containment challenged (5, 6) **CG1.1** RCS level **cannot** be monitored for ≥ 30 min. (Note 1) **AND** Core uncovery is indicated by **ANY** of the following: • UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT levels of sufficient magnitude to indicate core uncovery • Containment Radiation Monitor (RE 4596A or B) reading > 16 R/hr • Refueling Bridge Portable Area Radiation Monitor reading > 30 R/hr • Erratic Source Range Monitor indication **AND** **ANY** Containment Challenge indication, Table C-1 **Table C-1 Containment Challenge Indications** • CONTAINMENT CLOSURE **not** established (Note 6) • Containment Hydrogen concentration > 4% • UNPLANNED rise in Containment pressure	**CS1** Loss of RCS inventory affecting core decay heat removal capability (5, 6) **CS1.1** RCS level **cannot** be monitored for ≥ 30 min. (Note 1) **AND** Core uncovery is indicated by **ANY** of the following: • UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT levels of sufficient magnitude to indicate core uncovery • Containment Radiation Monitor (RE 4596A or B) reading > 16 R/hr • Refueling Bridge Portable Area Radiation Monitor reading > 30 R/hr • Erratic Source Range Monitor indication	**CA1** Loss of RCS inventory (5, 6) **CA1.1** Loss of RCS inventory as indicated by RCS level ≤ 0.4 ft (LI 10596) **CA1.2** RCS level **cannot** be monitored for ≥ 15 min. (Note 1) **AND EITHER** • UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT due to a loss of RCS inventory • Visual observation of UNISOLABLE RCS leakage	**CU1** UNPLANNED loss of RCS inventory for 15 minutes or longer (5, 6) **CU1.1** UNPLANNED loss of reactor coolant results in RCS level less than a required lower limit for ≥ 15 min. (Note 1) **CU1.2** RCS level **cannot** be monitored **AND EITHER:** • UNPLANNED increase in Containment Sumps, Auxiliary Building Sumps, BWST or RCDT due to a loss of RCS inventory • Visual observation of UNISOLABLE RCS leakage
2 Loss of Essential AC Power	None	**Table C-2 Offsite/Onsite AC Power Sources** **Offsite:** • X01 • X02 • X11 (back-fed via Main Transformer) **Onsite:** • EDG1 • EDG2 • SBODG	**CA2** Loss of **ALL** offsite power and **ALL** onsite AC power to essential buses for 15 minutes or longer (5, 6, DEF) **CA2.1** Loss of **ALL** offsite and **ALL** onsite AC power capability, Table C-2, to essential 4160V buses C1 and D1 for ≥ 15 min. (Note 1)	**CU2** Loss of **ALL** but one AC power source to essential buses for 15 minutes or longer (5, 6, DEF) **CU2.1** AC power capability, Table C-2, to essential 4160V buses C1 and D1 reduced to a single power source for ≥ 15 min. (Note 1) **AND** **ANY** additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS.
3 RCS Temp.	None	**Table C-3 RCS Reheat Duration Thresholds** * If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced the EAL is **not** applicable RCS Status / Containment Closure Status / Heat-up Duration Intact (but **not** reduced inventory) / N/A / 60 min. * **Not** intact **OR** At reduced inventory / established / 20 min. * **Not** intact **OR** At reduced inventory / **not** established / 0 min.	**CA3** Inability to maintain plant in cold shutdown (5, 6) **CA3.1** UNPLANNED increase in RCS temperature to > 200°F for > Table C-3 duration. (Note 1, 10) **OR** UNPLANNED RCS pressure increase > 10 psig due to a loss of RCS cooling (This EAL does not apply during water-solid plant conditions).	**CU3** UNPLANNED increase in RCS temperature (5, 6) **CU3.1** UNPLANNED increase in RCS temperature to > 200°F due to loss of decay heat removal capability. **CU3.2** Loss of **ALL** RCS temperature and RCS level indication for ≥ 15 min. (Note 1)
4 Loss of Essential DC Power	None	None		**CU4** Loss of essential DC power for 15 minutes or longer (5, 6) **CU4.1** < 105 VDC voltage indications on Technical Specification **required** essential 125 VDC distribution panels for ≥ 15 min. (Note 1)
5 Loss of Comm.	None	None	**Table C-4 Communications Methods** System / Onsite / ORO / NRC Public Address (Gaitronics) / X / / Onsite Radios / X / / Plant Telephones / X / X / X Commercial Telephones / X / X / X 4-Way Ringdown Circuit / / X / Satellite Phones / / X / X Cellular Phones / / X / X NRC Emergency Telephone System / / / X	**CU5** Loss of **ALL** onsite or offsite communications capabilities (5, 6, DEF) **CU5.1** Loss of **ALL** Table C-4 onsite communication methods. **CU5.2** Loss of **ALL** Table C-4 ORO communication methods. **CU5.3** Loss of **ALL** Table C-4 NRC communication methods.
6 Hazardous Event Affecting Safety Systems	None	**Table C-5 Hazardous Events** • Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Emergency Director	**CA6** Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode (5, 6) **CA6.1** The occurrence of **ANY** Table C-5 Hazardous Event **AND EITHER**: • Event has caused indications of degraded performance in at least one train of a SAFETY SYSTEM required for the current operating mode • The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure required for the current operating mode	None

Notes

Note 1: The Emergency Director 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 10: 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.

Enclosure B
L-16-020

EAL Calculations
(50 Pages Follow)

EP-EALCALC-DB-0701

Containment Radiation Monitor Readings Following Clad Damage (FC2 Loss, FC7 Loss, and CT2 Potential Loss)

(REVISION 0)

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Document Reviewer: Greg Van Wey

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Review:	Greg Van Wey	10/29/07 Date

TABLE OF CONTENTS

1. PURPOSE

Documentation of the assumptions, calculations and results for the Fission Product Barrier (FPB) threshold values for containment radiation monitor readings corresponding to a loss of fuel clad (FC2.1 Loss) and potential loss of containment (CT2.1 Potential Loss).

2. DISCUSSION

The Davis-Besse EAL Technical Basis Document contains the bases and references for the site specific EAL threshold values used to implement the NEI 99-01 Rev. 5 guidance methodology. This document has been developed to provide additional detailed technical documentation on how the containment radiation monitor values used to indicate a loss of the fuel clad and potential loss of containment fission product barriers were derived.

The generic guidance provided in NEI 99-01 is not intended to be used "as is." It is intended to give the logic for developing site specific EAL threshold values (Section 5.3 of NEI 99-01).

NEI 99-01 Rev. 5 provides the following guidance for FC2(L) as indicated by a containment radiation monitor reading:

> *The site specific reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment.*
>
> *The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 µCi/gm dose equivalent I-131 into the containment atmosphere.*
>
> *Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage.*
>
> *Caution: it is important to recognize that in the event the radiation monitor is sensitive to shine from the reactor vessel or piping, spurious readings will be present and another indicator of fuel clad damage is necessary or compensated for in the threshold value.*

It also provides the following guidance for CT2(PL) as indicated by a containment radiation monitor reading:

> *The site specific reading is a value which indicates significant fuel damage well in excess of the thresholds associated with both loss of Fuel Clad and loss of RCS barriers. As stated in Section 3.8* [of NEI 99-01], *a major release of radioactivity requiring off-site protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant.*
>
> *Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted.*
>
> *NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%. Unless there is a site specific analysis justifying a higher value, it is recommended that a radiation monitor reading corresponding to 20% fuel clad damage be specified here.*

The following calculations were performed to estimate the containment radiation monitor readings resulting for a release of the RCS inventory to the containment atmosphere as follows:

1) % clad damage associated with an RCS activity of 300 µCi/gm dose equivalent I-131.

2) Containment radiation monitor readings for various times after reactor shutdown associated with the clad damage equivalent of 300 µCi/gm dose equivalent I-131.

3) Containment radiation monitor readings for various times after reactor shutdown associated with 20% clad damage.

See section 3 for the assumptions, calculations and results.

3. DETERMINATION OF VALUES

3.1. Assumptions and Constants

3.1.1. 453.6 gm per lbm unit conversion factor.

3.1.2. RCS mass is 504,433 lbm at NOP, NOT, and normal pressurizer level (C-NSA-028.01-007 R0)

3.1.3. Source term activity is based on 24 month core activity values taken from USAR Table 15A-6.

3.1.4. DEI isotopic conversion factors are taken from DB-CH-01815.

3.2. Calculations

3.2.1. Fuel Clad Damage Estimate Based on 300 µCi/gm DEI-131

See Attachment 1 for the iterative results to the below calculation steps of the 24 month fuel cycle clad source term activity and the % clad damage.

1. 100% Fuel Clad Gap Activity

   USAR Table 15A-2 isotopic ratios are used to determine the Table 15A-6 equivalent clad gap activities are as follows:

   $15A\text{-}6_{Clad} = 15A\text{-}6_{Total} \times (15A\text{-}2_{Clad} / 15A\text{-}2_{Total})$

2. RCS Mass

   RCS Mass (gm) = RCS Mass (lbm) x 453.6 gm/lbm

   2.29E+08 gm = 504,433 lbm x 453.6 gm/lbm

3. Total Dose Equivalent I-131 Activity

   $DEI\ Act_{Total}$ (Ci) = DEI Conc (µCi/gm) x RCS Mass (gm) x 1E-6

   6.86E+04 Ci = 300 µCi/gm x 2.29E+08 gm x 1E-6 Ci/µCi

4. Fuel Clad Isotopic Activity Fractions (IAF)

   IAF_i = Fuel Clad $Activity_i$ / Fuel Clad $Activity_{Total}$

5. Normalized DEI Activity (NDA)

   $NDA_i = IAF_i \times DEI\ Factor_i$

6. Normalized DEI Activity Fractions (NDAF)

   $NDAF_i = NDA_i / NDA_{Total}$

7. Isotopic DEI Activity

   $DEI\ Activity_i = NDAF_i \times DEI\ Activity_{Total}$

8. Isotopic Adjusted Activity

   $Adjusted\ Activity_i = DEI\ Activity_i / DEI\ Factor_i$

9. % Clad Damage for 300 µCi/gm DEI-131

   $\%\ Clad\ Damage = Adjusted\ Activity_{Total} / 15A\text{-}6_{100\%\ Clad}$

   4.33% = 9.26E+04 Ci / 2.14E+06 Ci

3.2.2. Containment Radiation Monitor (CRM) Readings for Clad Damage

CRM readings in R/hr corresponding to 100% and 1% clad damage were derived from NUREG/BR-0150 Figure A-6 as follows:

Table 3.1		High (100%)	Low (1%)
Sprays Off			
	TAS = 1hr	7.00E+04	7.00E+02
	TAS = 24hr	2.00E+04	2.00E+02

CRM readings in R/hr corresponding to the 20% clad damage (CT2 potential loss) and 4.33% clad damage (FC2 loss) thresholds are as follows:

Table 3.2		High (20%)	Low (4.33%)
Sprays Off			
	TAS = 1hr	1.40E+04	3.03E+03
	TAS = 24hr	4.00E+03	8.66E+02

3.2.3. Curve Fit

A Microsoft Excel regression analysis[1] was run using Table 3.2 values to determine the slope and intercept of the CRM readings. The analysis is based on the equation 'x = b*m^y' (which provides an exponential trend). Results are as follows:

Table 3.3		High (20%)	Low (4.33%)
Sprays Off			
	m	0.9996822451	0.9985331536
	b	85.56491798	85.56491798

[1] **LOGEST(known_y's**,known_x's,const,stats)

From this, a predicted CRM reading can be determined for any time after shutdown as follows:

$$CRM = \frac{\ln\left(\frac{\mathrm{TAS}}{\mathrm{b}}\right)}{\ln(\mathrm{m})}$$

See Attachment 2 for results.

4. CONCLUSIONS

4.1. 300 µCi/gm DEI-131 is equivalent to 4.33% fuel clad (gap) damage

4.2. 4.33% fuel clad damage corresponds to a containment radiation monitor reading of 3.03E+03 R/hr at 1 hour and 8.66E+02 R/hr at 24 hours.

4.3. 20% fuel clad damage corresponds to a containment radiation monitor reading of 1.40E+04 R/hr at 1 hour and 4.00E+03 R/hr at 24 hours.

5. REFERENCES

5.1. NEI 99-01 R5, Methodology for Development of Emergency Action Levels

5.2. NUREG/BR-0150 Vol 1 R4, Response Technical Manual (RTM-96), March 1996

5.3. USAR Table 15A-6, Comparison of Core Fission Product Inventory for a 24 month Fuel Cycle with Source Terms from Table 15A-2 and TID 14844

5.4. C-NSA-028.01-007 R0, Control Room, LPZ, and EAB Radiation Doses due to ECCS leakage to the BWST and Auxiliary Building, 4/12/07.

5.5. DB-CH-01815, Dose Equivalent I-131 Determination, Revision 1, 2/7/05

6. ATTACHMENTS

6.1. 300 µCi/gm DEI-131 Equivalent Clad Damage

6.2. 20% and 4.33% CRM Readings vs. Time After Shutdown (TAS)

6.3. CRM vs. Time After Shutdown Graphs

24 Month Fuel Cycle Clad Source Term Activity

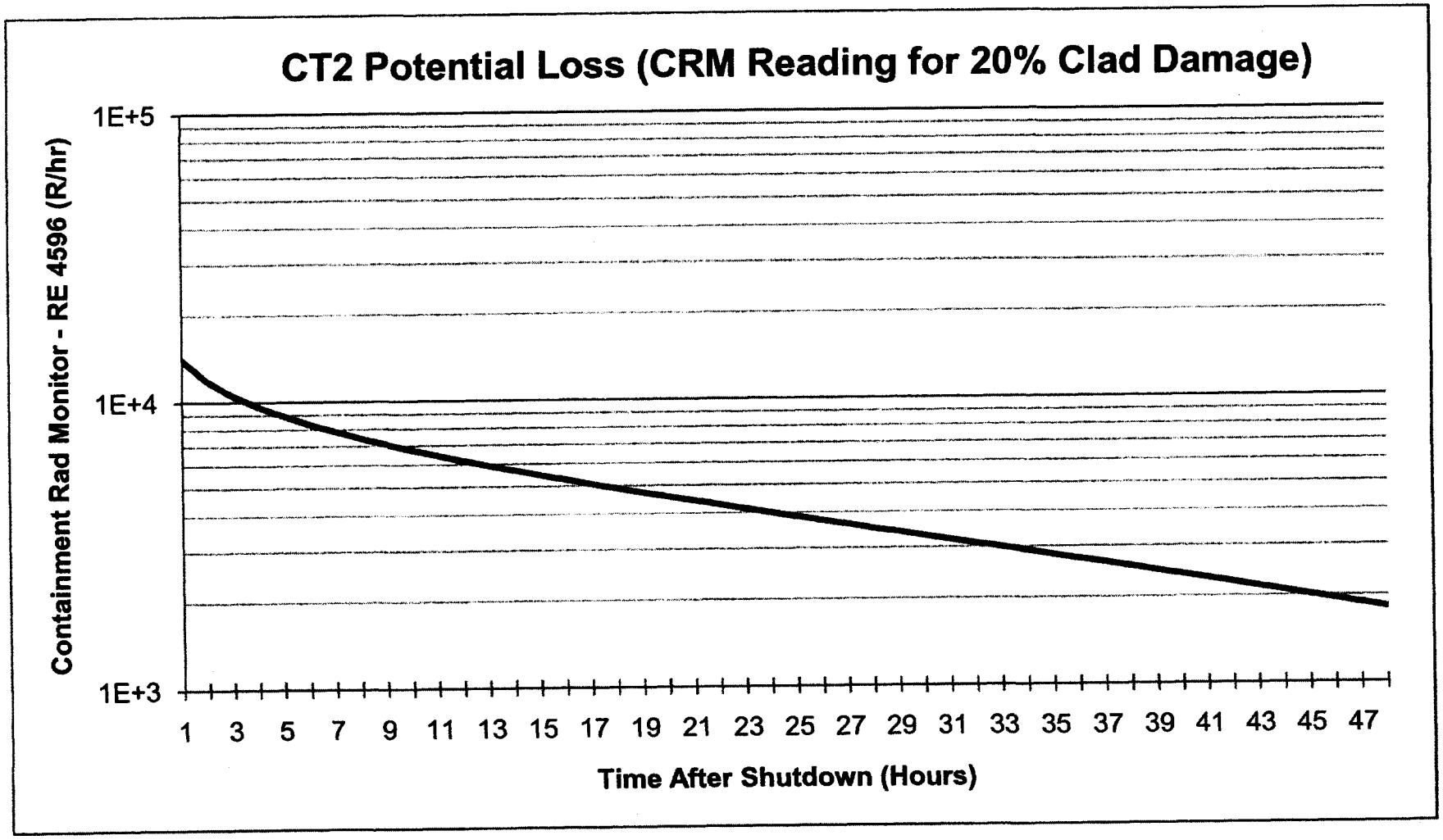
	USAR Table 15A-2			USAR Table 15A-6		
	Fuel	Clad	Total	Fuel	Clad	Total
I-131	7.46E+07	1.46E+06	7.61E+07	7.56E+07	1.48E+06	7.71E+07
I-132	1.09E+08	2.01E+05	1.09E+08	1.12E+08	2.06E+05	1.12E+08
I-133	1.44E+08	3.09E+05	1.44E+08	1.56E+08	3.34E+05	1.56E+08
I-134	1.83E+08	1.92E+04	1.83E+08	1.71E+08	1.79E+04	1.71E+08
I-135	1.40E+08	9.73E+04	1.40E+08	1.46E+08	1.01E+05	1.46E+08
Total	6.51E+08	2.09E+06	6.53E+08	6.60E+08	2.14E+06	6.62E+08

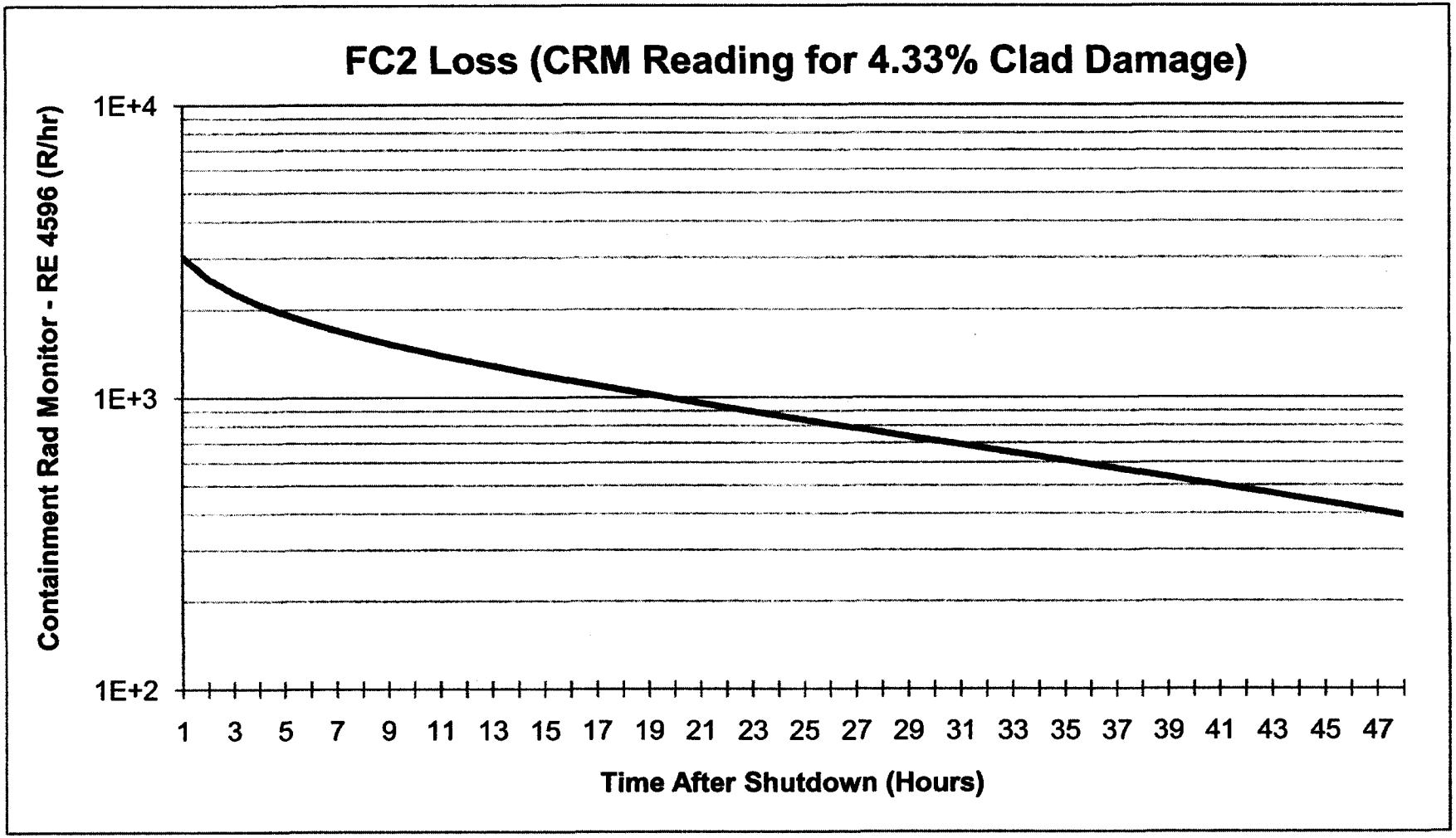
% Clad Damage Calculation

	Fuel Clad Activity (in Curies)	Isotopic Activity Fractions (IAF)	DEI Factor (DB-CH-01815)	IAF x DEI Factor = NDA (Normalized DEI Activity)	NDA Fraction (NDAF)	DEI Activity (in Curies)	Adjusted Activity (in Curies)
I-131	1.48E+06	69.17%	1.00E+00	6.92E-01	93.28%	6.40E+04	6.40E+04
I-132	2.06E+05	9.64%	3.62E-02	3.48E-03	0.47%	3.22E+02	8.92E+03
I-133	3.34E+05	15.61%	2.70E-01	4.22E-02	5.69%	3.91E+03	1.45E+04
I-134	1.79E+04	0.84%	1.69E-02	1.42E-04	0.02%	1.31E+01	7.76E+02
I-135	1.01E+05	4.74%	8.38E-02	3.97E-03	0.54%	3.68E+02	4.39E+03
Total	2.14E+06	100%	1.41E+00	7.42E-01	100%	6.86E+04	9.26E+04

RCS Mass (lbm):	504433
Conversion Factor (gm/lbm):	453.6
RCS Mass (gm):	2.29E+08
DEI-131 (uCi/gm):	300
Total DEI-131 (uCi):	6.86E+10
Total DEI-131 (Ci):	**6.864E+04**
	4.33%

	High (100%)	Low (1%)	CT2 (20%)	FC2 (4.33%)
m	0.9999364409	0.9936640497	0.9996822451	0.9985331536
b	85.56491798	85.56491798	85.56491798	85.56491798
TAS	100%	1%	20%	4.33%
1	7.00E+04	7.00E+02	1.40E+04	3.03E+03
2	5.91E+04	5.91E+02	1.18E+04	2.56E+03
3	5.27E+04	5.27E+02	1.05E+04	2.28E+03
4	4.82E+04	4.82E+02	9.64E+03	2.09E+03
5	4.47E+04	4.47E+02	8.94E+03	1.93E+03
6	4.18E+04	4.18E+02	8.36E+03	1.81E+03
7	3.94E+04	3.94E+02	7.88E+03	1.71E+03
8	3.73E+04	3.73E+02	7.46E+03	1.61E+03
9	3.54E+04	3.54E+02	7.09E+03	1.53E+03
10	3.38E+04	3.38E+02	6.75E+03	1.46E+03
11	3.23E+04	3.23E+02	6.45E+03	1.40E+03
12	3.09E+04	3.09E+02	6.18E+03	1.34E+03
13	2.96E+04	2.96E+02	5.93E+03	1.28E+03
14	2.85E+04	2.85E+02	5.70E+03	1.23E+03
15	2.74E+04	2.74E+02	5.48E+03	1.19E+03
16	2.64E+04	2.64E+02	5.28E+03	1.14E+03
17	2.54E+04	2.54E+02	5.09E+03	1.10E+03
18	2.45E+04	2.45E+02	4.91E+03	1.06E+03
19	2.37E+04	2.37E+02	4.74E+03	1.03E+03
20	2.29E+04	2.29E+02	4.57E+03	9.90E+02
21	2.21E+04	2.21E+02	4.42E+03	9.57E+02
22	2.14E+04	2.14E+02	4.27E+03	9.25E+02
23	2.07E+04	2.07E+02	4.13E+03	8.95E+02
24	2.00E+04	2.00E+02	4.00E+03	8.66E+02
25	1.94E+04	1.94E+02	3.87E+03	8.38E+02
26	1.87E+04	1.87E+02	3.75E+03	8.11E+02
27	1.81E+04	1.81E+02	3.63E+03	7.86E+02
28	1.76E+04	1.76E+02	3.51E+03	7.61E+02
29	1.70E+04	1.70E+02	3.40E+03	7.37E+02
30	1.65E+04	1.65E+02	3.30E+03	7.14E+02
31	1.60E+04	1.60E+02	3.19E+03	6.92E+02
32	1.55E+04	1.55E+02	3.09E+03	6.70E+02
33	1.50E+04	1.50E+02	3.00E+03	6.49E+02
34	1.45E+04	1.45E+02	2.90E+03	6.29E+02
35	1.41E+04	1.41E+02	2.81E+03	6.09E+02
36	1.36E+04	1.36E+02	2.72E+03	5.90E+02
37	1.32E+04	1.32E+02	2.64E+03	5.71E+02
38	1.28E+04	1.28E+02	2.55E+03	5.53E+02
39	1.24E+04	1.24E+02	2.47E+03	5.35E+02
40	1.20E+04	1.20E+02	2.39E+03	5.18E+02
41	1.16E+04	1.16E+02	2.31E+03	5.01E+02
42	1.12E+04	1.12E+02	2.24E+03	4.85E+02
43	1.08E+04	1.08E+02	2.17E+03	4.69E+02
44	1.05E+04	1.05E+02	2.09E+03	4.53E+02
45	1.01E+04	1.01E+02	2.02E+03	4.38E+02
46	9.76E+03	9.76E+01	1.95E+03	4.23E+02
47	9.43E+03	9.43E+01	1.89E+03	4.08E+02
48	9.09E+03	9.09E+01	1.82E+03	3.94E+02


CT2 Potential Loss (CRM Reading for 20% Clad Damage)
Containment Rad Monitor - RE 4596 (R/hr)
1E+5
1E+4
1E+3
1 3 5 7 9 11 13 15 17 19 21 23 25 27 29 31 33 35 37 39 41 43 45 47
Time After Shutdown (Hours)


FC2 Loss (CRM Reading for 4.33% Clad Damage)
Containment Rad Monitor - RE 4596 (R/hr)
1E+4
1E+3
1E+2
1 3 5 7 9 11 13 15 17 19 21 23 25 27 29 31 33 35 37 39 41 43 45 47
Time After Shutdown (Hours)

EP-EALCALC-DB-0702

Containment Radiation Monitor Reading Following a LOCA (RC2 Loss)

(REVISION 0)

Document Author: Scott McCain

Document Reviewer: Greg Van Wey

Author: Scott McCain — 10/17/07 Date

Review: Greg Van Wey — 10/29/07 Date

TABLE OF CONTENTS

1. PURPOSE

Documentation of the assumptions, calculations and results for the Fission Product Barrier (FPB) threshold value for a containment radiation monitor reading corresponding to a loss of RCS (RC2.1 Loss).

2. DISCUSSION

The Davis Besse EAL Technical Basis Document contains the bases and references for the site specific EAL threshold values used to implement the NEI 99-01 Rev. 5 guidance methodology. This document has been developed to provide additional detailed technical documentation on how the containment radiation monitor value used to indicate a loss of the RCS fission product barrier was derived.

The generic guidance provided in NEI 99-01 is not intended to be used "as is." It is intended to give the logic for developing site specific EAL threshold values (Section 5.3 of NEI 99-01).

NEI 99-01 Rev. 5 provides the following guidance for the loss of the RCS barrier as indicated by a containment radiation monitor reading:

> *The site specific reading is a value which indicates the release of reactor coolant to the containment.*
>
> *The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within T/S) into the containment atmosphere.*
>
> *This reading will be less than that specified for FC2(L)1. Thus, this threshold would be indicative of a RCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad barrier threshold, fuel damage would also be indicated.*
>
> *However, if the site specific physical location of the containment radiation monitor is such that radiation from a cloud of released RCS gases could not be distinguished from radiation from adjacent piping and components containing elevated reactor coolant activity, this threshold should be omitted and other site specific indications of RCS leakage substituted.*

The following calculations were performed to estimate the containment radiation monitor reading resulting from a release of the RCS inventory to the containment atmosphere using the USAR maximum RCS coolant activity (no core damage). See section 3 for the assumptions, calculations and results.

3. DETERMINATION OF VALUES

3.1. Assumptions and Constants

3.1.1. 28316.85 cc per ft^3 conversion factor.

3.1.2. Containment free volume is 2.834E+06 ft^3 (USAR 6.2.1.2.1)

3.1.3. Radius of the containment is 65 feet (Drawing M-127 R5)

3.1.4. RCS liquid volume is 11380 ft^3 at NOP, NOT, and normal pressurizer level of 220 inches (DB-NE-06201)

3.1.5. Maximum RCS activity in µCi/cc is taken from USAR Table 15A-4.

3.2. Calculations

See Attachment 1 for the results to the calculations of the available isotopic activity.

3.3. MicroShield Modeling

3.3.1. Model Geometry

Geometry 8, cylinder volume with end shields was selected as the basis MicroShield model for the calculation (end shielding was not modeled to provide dose rate reduction).

Cylinder height was determined by the following equation to achieve the desired free volume:

$$\mathbf{h = Cont\ Vol / \pi r^2}$$

$$213.5 = 2.834E{+}06 / \pi 65^2$$

Drawing M-127 R5A indicates a height of 215 feet to the top of the containment dome which verifies the use of 213.5 feet as an acceptable approximation for the calculation geometry.

3.3.2. Dose Receptor

The MicroShield dose receptor representing the containment radiation monitor was positioned at 55 feet along the x-axis. MicroShield modeling limitations do not allow the receptor point to be placed 40 feet up the cylinder wall (el. 643 feet) from the base (el. 603 feet). This limitation is not considered significant with regard to the overall accuracy of the calculation.

3.4. Results

MicroShield calculation results provide a CRM dose rate of 1.58E+04 mR/hr. The detailed MicroShield report is provided in Attachment 2.

4. CONCLUSIONS

RE 4596 A and B are downscale during normal power operations and indicate approximately 1.5 R/hr from their trace signal. The maximum RCS activity calculation results provide a value about 10 times the downscale reading. It is also considerably lower that the Fuel Clad loss value of 866 R/hr @1 hour, which provides consistency with the development guidance of NEI 99-01.

Projected Readings from the Release of Maximum RCS Inventory vs. Normal Operational Readings

Projected Dose Rate	CRM Operational Dose Rate	Recommended Value
1.58E+04 mR/hr	~ 1.5 R/hr	15 R/hr

A value of 15 R/hr has been selected as the RC2.1 loss threshold.

5. REFERENCES

5.1. NEI 99-01 R5, Methodology for Development of Emergency Action Levels

5.2. USAR 6.2.1.2.1, (Containment) Design Parameters

5.3. USAR Table 15A-4, Maximum Fission Product Activity in Reactor Coolant

5.4. Drawing M-127 R5, Equipment Locations Containment Vessel Section "A" – "A"

5.5. SD-17B R4 Table 2.3-1, Area Radiation Monitor Location.

5.6. Procedure DB-NE-06201, Reactor Operators Curve Book, Revision 9.

6. ATTACHMENTS

6.1. RCS Maximum Isotopic Activity

6.2. MicroShield CRM Calculation Report

Conversion Factor cc/ft3):	28316.85
RCS Volume (ft3):	1.138E+04
RCS Volume (cc):	3.222E+08
Containment Volume (ft3):	2.834E+06
Containment Volume (cc):	8.025E+10

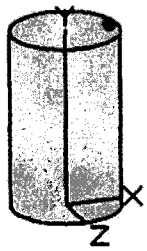
	RCS Concentration (uCi/cc)	Available Isotopic Activity (Ci)	Cont Concentration (uCi/cc)
Kr 83m	3.12E-01	1.0054E+02	1.25E-03
Kr 85	9.30E+00	2.9969E+03	3.73E-02
Kr 85m	1.65E+00	5.3171E+02	6.63E-03
Kr 87	9.04E-01	2.9131E+02	3.63E-03
Kr 88	2.90E+00	9.3451E+02	1.16E-02
Xe-131m	2.32E+00	7.4761E+02	9.32E-03
Xe-133	2.63E+02	8.4751E+04	1.06E+00
Xe-133m	3.03E+00	9.7640E+02	1.22E-02
Xe-135	5.50E+00	1.7724E+03	2.21E-02
Xe-135m	1.00E+00	3.2225E+02	4.02E-03
Xe-138	5.52E-01	1.7788E+02	2.22E-03
I-131	3.48E+00	1.1214E+03	1.40E-02
I-132	5.20E+00	1.6757E+03	2.09E-02
I-133	4.11E+00	1.3244E+03	1.65E-02
I-134	5.39E-01	1.7369E+02	2.16E-03
I-135	2.05E+00	6.6060E+02	8.23E-03

MicroShield v5.03 (5.03-00091)
Toledo Edison Company

Page : 1
DOS File : EP-L-RC2.MS5
Run Date : October 3, 2007
Run Time : 9:51:35 AM
Duration : 00:00:02

File Ref: _____
Date: _____
By: _____
Checked: _____

Case Title: RC2 Loss CRM Reading
Description: CRM Reading Based on Max RCS Activity in Containment
Geometry: 8 - Cylinder Volume - End Shields



Source Dimensions

Height	6.5e+3 cm	213 ft 6.0 in
Radius	2.0e+3 cm	65 ft 0.0 in

Dose Points

	X	Y	Z
# 1	1676.4 cm	6.51e+03 cm	0 cm
	55 ft 0.0 in	213 ft 6.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Source	8.02e+10 cm³	Air	0.00122
Air Gap		Air	0.00122

Source Input
Grouping Method : Standard Indices
Number of Groups : 25
Lower Energy Cutoff : 0.015
Photons < 0.015 : Excluded
Library : Grove

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
I-131	1.1214e+003	4.1492e+013	1.3975e-002	5.1706e+002
I-132	1.6757e+003	6.2001e+013	2.0882e-002	7.7264e+002
I-133	1.3244e+003	4.9003e+013	1.6504e-002	6.1066e+002
I-134	1.7369e+002	6.4265e+012	2.1645e-003	8.0086e+001
I-135	6.6060e+002	2.4442e+013	8.2323e-003	3.0459e+002
Kr-83m	1.0054e+002	3.7200e+012	1.2529e-003	4.6358e+001
Kr-85	2.9969e+003	1.1089e+014	3.7347e-002	1.3818e+003
Kr-85m	5.3171e+002	1.9673e+013	6.6261e-003	2.4516e+002
Kr-87	2.9131e+002	1.0778e+013	3.6302e-003	1.3432e+002
Kr-88	9.3451e+002	3.4577e+013	1.1646e-002	4.3089e+002
Xe-131m	7.4761e+002	2.7662e+013	9.3166e-003	3.4471e+002
Xe-133	8.4751e+004	3.1358e+015	1.0561e+000	3.9078e+004
Xe-133m	9.7640e+002	3.6127e+013	1.2168e-002	4.5020e+002
Xe-135	1.7724e+003	6.5579e+013	2.2087e-002	8.1723e+002
Xe-135m	3.2225e+002	1.1923e+013	4.0158e-003	1.4859e+002
Xe-138	1.7788e+002	6.5816e+012	2.2167e-003	8.2018e+001

Buildup
The material reference is : Source

Page : 2
DOS File : EP-L-RC2.MS5
Run Date : October 3, 2007
Run Time : 9:51:35 AM
Duration : 00:00:02

Integration Parameters

Radial	20
Circumferential	10
Y Direction (axial)	10

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.03	1.548e+15	3.322e+05	5.419e+05	3.293e+03	5.370e+03
0.08	1.152e+15	8.634e+05	1.618e+06	1.366e+03	2.560e+03
0.1	6.819e+10	6.515e+01	1.129e+02	9.967e-02	1.727e-01
0.15	1.752e+13	2.592e+04	4.031e+04	4.268e+01	6.638e+01
0.2	7.508e+13	1.514e+05	2.124e+05	2.673e+02	3.749e+02
0.3	1.040e+13	3.248e+04	4.199e+04	6.162e+01	7.965e+01
0.4	4.616e+13	1.966e+05	2.423e+05	3.830e+02	4.721e+02
0.5	7.146e+13	3.869e+05	4.621e+05	7.594e+02	9.071e+02
0.6	8.762e+13	5.769e+05	6.743e+05	1.126e+03	1.316e+03
0.8	8.249e+13	7.391e+05	8.380e+05	1.406e+03	1.594e+03
1.0	3.527e+13	4.010e+05	4.466e+05	7.392e+02	8.233e+02
1.5	3.141e+13	5.495e+05	5.955e+05	9.245e+02	1.002e+03
2.0	2.934e+13	6.951e+05	7.409e+05	1.075e+03	1.146e+03
3.0	1.774e+12	6.427e+04	6.736e+04	8.719e+01	9.139e+01
TOTALS:	3.189e+15	5.015e+06	6.522e+06	1.153e+04	1.580e+04

Radiological Gaseous Effluent EAL Values
(EALs RG1, RS1, RA1 and RU1)

(REVISION 2)

Document Author: Gregory S. Van Wey

Document Reviewer: Vince Capozziello

Author: _____ 11/17/15
Gregory S. Van Wey — Date

Review: _____ 11/17/15
Date

TABLE OF CONTENTS

1. PURPOSE

The DBNPS Emergency Action Level (EAL) Technical Bases Manual contains background information, event declaration thresholds, bases and references for the site specific EAL and Fission Product Barrier (FPB) values used to implement the Nuclear Energy Institute (NEI) Rev. 6 EAL guidance methodology. This calculation document provides additional technical detail specific to the derivation of the gaseous and liquid radiological effluent EAL values developed in accordance with the guidance in NEI 99-01, Rev. 6.

Documentation of the assumptions, calculations and results are provided for the DBNPS site specific Rx1 series EAL effluent monitor values associated with the NEI 99-01, Rev. 6 EALs listed below:

- NEI AU1.1 (gaseous and liquid)
- NEI AA1.1 (gaseous and liquid)
- NEI AS1.1 (gaseous)
- NEI AG1.1 (gaseous)

2. DEVELOPMENT METHODOLOGY AND BASES

2.1 Threshold Limits

2.1.1 RU1.1 Liquid Threshold Limits

Guidance Criteria

The RU1 Initiating Condition (IC) addresses a release of radioactivity that, for whatever reason, causes effluent radiation monitor readings to exceed 2 times the Offsite Dose Calculation Manual (ODCM) limit for 60 minutes or longer.

DBNPS Bases

The DBNPS liquid effluent limits for RU1 are taken from the ODCM, Liquid Effluent Monitor Setpoints, Section 2.3.3.

- For individual nuclides, the limit is the concentration value specified in 10CFR20, Appendix B, Table 2, Column 2.
- For dissolved and entrained noble gases, the limit is 2E-4 μCi/ml total activity.
- For nuclide mixtures, the limit is the summation of individual nuclide concentrations divided by their corresponding individual effluent concentration (EC) where the resulting value equals 1 (ODCM 2.3.3).

The DBNPS RU1.1 liquid effluent EAL threshold values will be established as the High Alarm level for the monitor and will equate to 2 times the ODCM limit.

2.1.2 RU1.1 Gaseous Threshold Limits

Guidance Criteria

The RU1 Initiating Condition (IC) addresses a release of radioactivity that, for whatever reason, causes effluent radiation monitor readings to exceed 2 times the Offsite Dose Calculation Manual (ODCM) limit for 60 minutes or longer.

DBNPS Bases

The DBNPS gaseous effluent limits for RU1.1 are based on values that equate to effluent radiation monitor readings exceeding 2 times the ODCM limit for 60 minutes or longer.

The DBNPS gaseous effluent limits for RU1.1 are taken from ODCM Section 3.3.2.

Note – Inhalation (internal organ) limits are not applicable for EAL threshold determination since the specified surveillance involves collection and analysis of composite samples. The after-the-fact assessment (individual uptake) could not be made in a timely manner conducive to accident classification.

- Less than or equal to 500 mrem/yr to the total (whole) body from noble gases
- Less than or equal to 3,000 mrem/yr to the skin from noble gases.

2.1.3 RA1.1 Liquid Threshold Limits

Guidance Criteria

The RA1 IC addresses a release of radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.

This is based on values at 1% of the EPA Protective Action Guides (PAGs).

Per NEI 99-01, the effluent monitor readings should correspond to the above dose limit at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.

DBNPS Bases

The DBNPS liquid effluent limits are based on the liquid effluent concentration values given in 10CFR20 Appendix B Table 2 Column 2 (see Section 2.2.1 above). The 10CFR20 values are equivalent to the radionuclide concentrations which, if ingested continuously over the course of a year, would produce a total effective dose equivalent of 0.05 rem (50 millirem). The EPA PAGs are based on a TEDE dose from immersion, inhalation and deposition. The 10CFR20 limits and the EPA limits do not represent the same type of exposure and thus cannot be compared on a one to one basis.

Additionally, significant dilution assumptions are incorporated in determining ODCM ingestion limits for liquid releases such that obtaining a dose of 10 mrem in one hour would require a discharge concentration above the effluent monitor threshold (ingestion of radioactivity from a liquid release at the site boundary is not practical).

Thus, DBNPS EALs will not contain a liquid effluent monitor threshold value that equates to 1% of the EPA PAG.

2.1.4 RA1.1 Gaseous Threshold Limits

Guidance Criteria

The RA1 IC addresses a release of radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.

Per NEI 99-01, the effluent monitor readings are based on values at 1% of the EPA Protective Action Guides (PAGs) at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.

The EPA PAG guidance provides for the use of adult thyroid dose conversion factors; however, some states have decided to base protective actions on child thyroid CDE. Nuclear power plant ICs/EALs need to be consistent with the protective action methodologies employed by the States within their EPZs. They thyroid CDE dose used in the IC and EALs should be adjusted as necessary to align with State protective action decision-making criteria.

DBNPS Bases

The DBNPS gaseous effluent limits for RA1.1 are based on values that equate to an offsite dose greater than 10 mrem TEDE or 50 mrem child thyroid CDE, which are effectively 1% of the EPA Protective Action Guides (PAGs) as the State of Ohio utilizes child thyroid CDE for protective action decision making.

2.1.5 RS1.1 Gaseous Threshold Limits

Guidance Criteria

The RS1 IC addresses a release of radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.

This is based on values at 10% of the EPA Protective Action Guides (PAGs) at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.

The EPA PAG guidance provides for the use of adult thyroid dose conversion factors; however, some states have decided to base protective actions on child thyroid CDE. Nuclear power plant ICs/EALs need to be consistent with the protective action methodologies employed by the States within their EPZs. They

thyroid CDE dose used in the IC and EALs should be adjusted as necessary to align with State protective action decision-making criteria.

DBNPS Bases

The DBNPS gaseous effluent limits for RS1.1 are based on values that equate to an offsite dose greater than 100 mrem TEDE or 500 mrem child thyroid CDE, which are effectively 10% of the EPA Protective Action Guides (PAGs) as the State of Ohio utilizes child thyroid CDE for protective action decision making.

2.1.6 RG1.1 Gaseous Threshold Limits

Guidance Criteria

The RG1 IC addresses a release of radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.

This is based on values at 100% of the EPA Protective Action Guides (PAGs) at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.

The EPA PAG guidance provides for the use of adult thyroid dose conversion factors; however, some states have decided to base protective actions on child thyroid CDE. Nuclear power plant ICs/EALs need to be consistent with the protective action methodologies employed by the States within their EPZs. The thyroid CDE dose used in the IC and EALs should be adjusted as necessary to align with State protective action decision-making criteria.

DBNPS Bases

The DBNPS gaseous effluent limits for RG1.1 are based on values that equate to an offsite dose greater than 1,000 mrem TEDE or 5,000 mrem child thyroid CDE, which are effectively 100% of the EPA Protective Action Guides (PAGs) as the State of Ohio utilizes child thyroid CDE for protective action decision making.

2.2 Effluent Release Points

2.2.1 DBNPS Liquid Release Points

Guidance Criteria

Per NEI 99-01, the RU1 IC addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways and planned batch releases from non-continuous release pathways.

Per NEI 99-01, the RA1 IC includes events or conditions involving a radiological release, whether gaseous or liquid, monitored or un-monitored. Classification based on effluent monitor readings assumes that a release path to the environment is established. If the associated release path to the environment

has been isolated, the effluent monitor reading is no longer VALID for emergency classification purposes.

The "site-specific monitor list and threshold values" should include the effluent monitors described in emergency plan and emergency dose assessment procedures.

DBNPS Bases

Batch Releases (ODCM Section 2.1.1.a) – Batch releases from the liquid radwaste system may occur from:

Point	Radiation Monitor	Function
Clean Radwaste Effluent Monitor Tanks (2)	RE-1770 A & B	Alarm/Automatic Release Termination
Miscellaneous Liquid Waste Monitor Tank	RE-1878 A & B	Alarm/Automatic Release Termination
Detergent Waste Drain Tank	RE-1878 A & B	Alarm/Automatic Release Termination

Continuous Releases (ODCM Section 2.1.2) – DBNPS does not perform continuous radioactive liquid releases. Potential leakage pathways identified in the ODCM that could result in continuous leakage to the environment should barrier systems fail are:

System	Radiation Monitor	Function
Storm Sewer Drain Line	RE-4686	Alarm only
Component Cooling Water System (CCWS)	RE-1412 & 1413	Alarm only
Service Water System (SWS)	RE-8432	Alarm only
Intake Forebay	RE-8434	Alarm only

The above pathways include installed monitors (not required by the ODCM) or are checked by periodic grab sample. These sources are not engineered discharge pathways used for normally occurring continuous liquid radioactivity releases and thus are not applicable for use as a routine discharge liquid release point for EAL threshold development.

2.2.2 DBNPS Gaseous Release Point

Guidance Criteria

Per NEI 99-01, the RU1 and RA1 ICs address normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways and planned batch releases from non-continuous release pathways.

Per NEI 99-01, the RS1 and RG1 ICs include events or conditions involving a radiological release, whether gaseous of liquid, monitored or un-monitored. Classification based on effluent monitor readings assumes that a release path to the environment is established. If the associated release path to the environment has been isolated, the effluent monitor reading is no longer VALID for emergency classification purposes.

The "site-specific monitor list and threshold values" should include the effluent monitors described in the emergency plan and emergency dose assessment procedures.

DBNPS Bases

All releases of gaseous radioactive effluents are designed to occur via the Station Vent (ODCM Section 3.3.2).

Note - Site ground level elevation is 585'0". The DBNPS primary release point (Station Vent) is considered ground level (USAR 11A.3.1)

2.3 Gaseous Thresholds Source Terms

2.3.1 RU1.1 Gaseous Threshold Source Term

Guidance Criteria

Per NEI 99-01, radiation monitor readings should reflect values that correspond to a radiological release exceeding 2 times a release control limit. The controlling document typically describes methodologies for determining effluent radiation monitor setpoints; these methodologies should be used to determine EAL values.

For EAL #2 (RU1.1) values should be 2 times the setpoint established by the radioactivity permit to warn of a release that is not in compliance with the specified limits. Indexing the value in this manner ensures consistency between the EAL and the setpoint established by a specific discharge permit.

DBNPS Bases

Radiation monitor setpoints are determined for each batch release based on the actual radionuclide mix that is contained in the tank prior to the release. For conservatism in determining the effluent radiation monitor values for this IC, the source term values are based on the estimated annual releases from gaseous radwaste systems, specifically the containment vessel purge, aux building and the steam jet air ejector, contained in USAR Table 11.3-14.

2.3.2 RA1.1, RS1.1, and RG1.1 Gaseous Thresholds Source Terms

Guidance Criteria

The calculation of monitor readings will require use of an assumed release isotopic mix; the selected mix should be the same for ICs AA1, AS1, and AG1.

DBNPS Bases

The accident source term in the MIDAS dose model is based on NUREG-1228, Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents, and RTM-96, Response Technical Manual, adjusted to DBNPS current core and plant configuration. The following MIDAS menu selections are used to establish the accident source term:

Accident: Containment Bypass
Core Condition: Gap Release, Uncovered 15-30 minutes
Filtration: Filtered

No credit is taken for source term decay. The start of release time entered into MIDAS is coincident with the time of reactor trip.

2.4 Release Duration

Guidance Criteria

Per NEI 99-01[1], since doses are generally monitored in real-time, it is suggested a release duration of one hour be assumed...If individual site analyses indicate a longer or shorter duration for the period in which the substantial portion of the activity is release, the longer duration should be used.

[1] Release duration basis is taken from NEI 99-01, Rev. 5, as NEI 99-01, Rev. 6 does not provide specific guidance for this input parameter.

DBNPS Bases

The DBNPS effluent monitor readings are calculated for a release duration of one hour. As a significant number of accident types, plant condition and operator action variables exist that would influence the duration of an effluent release, no

site specific analysis has been performed to establish a shorter or longer release duration.

2.5 Meteorology

Guidance Criteria

The effluent monitor readings should correspond to the applicable dose limit at the "site-specific dose receptor point". The "site-specific dose receptor point" is the distance(s) and/or locations used by the licensee to distinguish between on-site and offsite doses. The selected distance(s) and/or locations should reflect the content of the emergency plan, and the procedural methodology used to determine offsite doses and protective action recommendations. This is typically the boundary of the Owner Controlled Area.

DBNPS Bases

The DBNPS site specific meteorology used for the calculation of monitor readings for these ICs/EALs is based on annual average meteorology. This is considered acceptable in that dose assessments using actual meteorology will be initiated for significant radioactivity releases. The annual average meteorology inputs for these calculations are:

2.5.1 Wind speed

The average wind speed at the 10 meter level for DBNPS is 9.0 miles per hour (derived from historic Radiological Effluent Monitoring Program – REMP files and the OSIsoft PI server archives for the period from 1/1/97 to 12/13/06).

2.5.2 Stability Class

The most predominant stability for DBNPS is class "D" (derived from historic REMP files and the OSIsoft PI server archives for the period from 1/1/97 to 12/13/06).

2.5.3 Site Boundary distance used for calculations

For the RU1.1 calculations, the DBNPS "site-specific dose receptor point" utilized in the derivation of the EAL effluent release threshold, is the limiting receptor location for the downwind unrestricted area, or site boundary, which is 730 meters (0.45 miles) (1609.3 m/mi. conversion factor used) in the NNE direction. (ODCM Table 3-6 from the Chesapeake Nuclear Services report, "Davis-Besse Nuclear Power Station Meteorological and Atmospheric Dispersion Report, Rev. 1" dated June 2012).

For the RA1.1, RS1.1, and RG1.1 calculations, an average site boundary distance of 0.75 miles is used, as agreed to with the State of Ohio (CR 07-26574).

2.5.4 Atmospheric Dispersion

For RU1.1, methodology used in the ODCM is followed. The limiting receptor location for the downwind unrestricted area, or site boundary, is 730 meters (0.45 miles) in the NNE direction. The atmospheric dispersion used for this calculation is 5.90E-6 sec/m^3 X/Q value (ODCM Table 3-6 from the Chesapeake Nuclear Services report, "Davis-Besse Nuclear Power Station Meteorological and Atmospheric Dispersion Report, Rev. 1" dated June 2012).

For RA1.1, RS1.1, and RG1.1, calculations are performed using the MIDAS dose assessment software. The MIDAS software models atmospheric dispersion consistent with the regulatory guidance contained in Reg. Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants; Reg. Guide 1.23, Meteorological Monitoring Programs for Nuclear Power Plants; and Reg. Guide 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR50, Appendix I.

3. DESIGN INPUTS

3.1 Constants and Conversion Factors

3.1.1 1609.3 meters per mile (x value in the dispersion equations)

3.1.2 0.447 m/sec per mph

3.1.3 365.25 days per year

3.1.4 472 cc/sec per cfm

3.1.5 10^6 µCi per Ci

3.2 Station Vent Effluent Monitor and Flow Specifications

3.2.1 Station Vent Radiation Monitor (USAR Table 11.4-1)

- RE-4598 AAC/BAC (noble gas).....1E-7 to 1E-1 µCi/cc
- RE-4598 ABC/BBC (noble gas).....1E-4 to 1E5 µCi/cc

3.2.2 Normal Operations Ventilation Flow

The RU1.1 calculations are based on a normal operations station vent flow rate value of 1.21E+5 SCFM as follows:

- Main Station Exhaust (single train).....7.1E+04 SCFM

  USAR Table 11.4-1 ventilation flow rate (SCFM) for the RE-4598 detectors is provided as a band from 6.3E+04 to 7.1E+04.

- Containment Vessel Purge.....5.0E+04 SCFM

  USAR Table 11.3-6

- Steam Jet Air Ejectors.....1.0E+01 SCFM

  Normal average operational value

3.2.3 Emergency Operations Ventilation Flow

The RS1.1 and RG1.1 EAL calculations are based on an emergency station vent flow rate value of 8.7E+04 SCFM as follows:

- Main Station Exhaust (single train).....7.1E+04 SCFM

  USAR Table 11.4-1 ventilation flow rate (SCFM) for the RE-4598 detectors is provided as a band from 6.3E+04 to 7.1E+04.

- Emergency Ventilation System (two trains).....1.6E+04 SCFM

  USAR 11.3.2.2.4.c specifies EVS flow rate as 8.0E+03 SCFM per train.

4. CALCULATIONS

4.1 RU1.1 Gaseous Effluent Monitor value

4.1.1 Maximum ODCM Effluent Monitor Limit

1. Total Body Dose (DDE)

$$SP_{TB} = \frac{\sum C_i \times 500}{472 \times \frac{\chi}{Q_{AVG}} \times VF \times \sum(C_i \times K_i)}$$

2. Skin Dose (SDE)

$$SP_{SKIN} = \frac{\sum C_i \times 3000}{472 \times \frac{\chi}{Q_{AVG}} \times VF \times \sum(C_i \times (L_i + 1.1M_i))}$$

Where:

SP_x	Provides radiation monitor setpoint reading for given ODCM limit
C_i	Activity Released
Limit	ODCM Limit – 500 total body or 3000 skin (mRem/yr)
472	Conversion factor (CFM to cc/sec)
X/Q_{AVG}	Annual Average Dispersion Factor (sec/m^3)
VF	Vent Flow (CFM)
$\Sigma C_i x$...	Activity Adjusted Dose Correction Factor

See Attachment 1 for the ODCM setpoint spreadsheet calculations.

2.86E-3 µCi/cc.....$SP_{Total\ Body}$

8.28E-3 µCi/cc.....SP_{Skin}

4.1.2 RU1.1 Gaseous Effluent Monitor value

The RU1.1 value is 2 times the calculated ODCM effluent monitor setpoint for the Total Body (the Skin limit required a higher release rate to achieve the limit).

RU1.1 = 2 x SP_{TB}

See Attachment 1 for the actual RU1.1 spreadsheet calculations.

5.72E-3 µCi/cc.....**RU1.1**

4.2 RA1.1, RS1.1, RG1.1 Gaseous Effluent Monitor values

See Attachment 2 for the results of the MIDAS gaseous effluent calculations for the EAL threshold values for RA1.1, RS1.1, and RG1.1.

5. CONCLUSIONS

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	Station Vent (noble gas)	RE 4598 AB/BB	**8.4E +00** (µCi/cc)	**8.4E-01** (µCi/cc)	**8.4E-02** (µCi/cc)	-----
	Station Vent (noble gas)	RE 4598 AA/BA	-----	-----	-----	**5.72E-03** (µCi/cc)
	Waste Gas System Outlet	RE 1822 A/B	-----	-----	-----	**2 x High Alarm**
Liquid	Clean Waste System Outlet	RE 1770 A/B	-----	-----	-----	**2 x High Alarm**
	Misc. Waste System Outlet	RE 1878 A/B	-----	-----	-----	**2 x High Alarm**

6. REFERENCES

6.1. NEI 99-01 R6, Methodology for Development of Emergency Action Levels

6.2. Davis-Besse Offsite Dose Calculation Manual (ODCM), Revision 31

6.3. Davis-Besse Updated Safety Analysis Report (USAR), Revision 30

6.4 MIDAS version 1.5.16.101714

7. ATTACHMENTS

7.1. Attachment 1 - ODCM Limit Calculation

7.2. Attachment 2 – MIDAS Gaseous Effluent EAL Calculations

Nuclide	ODCM Table 3-5: Total Body Gamma Dose Factor Ki (mrem/yr per uCurie/m3)	ODCM Table 3-5: Skin Beta Dose Factor Li (mrem/yr per uCurie/m3)	ODCM Table 3-5: Gamma Air Dose Factor Mi (mrad/yr per uCuries/m3)	USAR Table 11.3-14: Containment Purge (Curies/yr)	USAR Table 11.3-14: Aux Building Ventilation (Curies/yr)	USAR Table 11.3-14: Steam Jet Air Ejector Vent (Curies/yr)	USAR Table 11.3-14: Total Annual Release (Curies/yr)	Relative Fraction	Ci x Ki	Ci x (Li + 1.1Mi)
Kr-83m	7.56E-02	0.00E+00	1.93E+01	2.49E-02	6.35E-01	3.27E+00	3.93E+00	9.96E-04	7.53E-05	2.11E-02
Kr-85m	1.17E+03	1.46E+03	1.23E+03	1.83E-01	3.37E+00	1.71E+01	2.06E+01	5.22E-03	6.11E+00	1.47E+01
Kr-85	1.61E+01	1.34E+03	1.72E+01	4.35E+01	6.93E+00	3.49E+01	8.54E+01	2.16E-02	3.48E-01	2.94E+01
Kr-87	5.92E+03	9.73E+03	6.17E+03	6.14E-02	1.84E+00	9.30E+00	1.12E+01	2.84E-03	1.68E+01	4.69E+01
Kr-88	1.47E+04	2.37E+03	1.52E+04	2.67E-01	5.89E+00	2.98E+01	3.60E+01	9.11E-03	1.34E+02	1.74E+02
Xe-131m	9.15E+01	4.76E+02	1.56E+02	6.17E+00	4.96E+00	2.49E+01	3.60E+01	9.13E-03	8.35E-01	5.91E+00
Xe-133m	2.51E+02	9.94E+02	3.27E+02	1.70E+00	6.25E+00	3.14E+01	3.94E+01	9.97E-03	2.50E+00	1.35E+01
Xe-133	2.94E+02	3.06E+02	3.53E+02	3.15E+02	5.47E+02	2.75E+03	3.61E+03	9.15E-01	2.69E+02	6.35E+02
Xe-135m	3.12E+03	7.11E+02	3.36E+03	2.74E-02	2.04E+00	1.03E+01	1.24E+01	3.13E-03	9.78E+00	1.38E+01
Xe-135	1.81E+03	1.86E+03	1.92E+03	8.62E-01	1.12E+01	5.65E+01	6.85E+01	1.74E-02	3.14E+01	6.90E+01
Xe-138	8.83E+03	4.13E+03	9.21E+03	1.59E-02	1.13E+00	5.70E+00	6.85E+00	1.73E-03	1.53E+01	2.47E+01
Ar-41	8.84E+03	2.69E+03	9.30E+03	9.38E-02	2.44E+00	1.23E+01	1.48E+01	3.75E-03	3.31E+01	4.84E+01
				3.68E+02	**5.93E+02**	**2.99E+03**	**3.95E+03**	**1.00E+00**	**5.19E+02**	**1.08E+03**

Calculation Constants

Total Body (DDE):	500	mRem/yr
Skin Dose (SDE):	3000	mRem/yr
Vent Flow:	1.21E+05	CFM
X/Q:	5.90E-06	sec/m3
Unit Conversion:	472	ml/ft3 x min/sec
Dose Conversion:	1.1	mRem/mRad

Calculation Results

SP (DDE):	2.86E-03	uCi/ml
SP (SDE):	8.28E-03	uCi/ml

2 times
5.72E-03

Dose Assessment Data Form

Insufficient Data to Determine Lake Breeze

SITE: DB ACTUAL MENU: B

Summary For Calculation: 11/14/15 14:00 DATE PRINTED: 11/14/15 13:56

Current Input Data

Meteorological Data:	Level	Wind Direction from	Adjusted Wind Speed	Measured Wind Speed	Precipitation inch/15 min	Stability Class
	Lower	90 degrees	9.0 mph	9.0 mph	No Rain	D

Monitor ID	Release Point	Monitor Reading	Monitor Status*	Monitor Units	Conversion to uCi/cc	Effective Concentration (uCi/cc)	Flow Reading	Flow Status*	Flow Units	Release Rate (uCi/sec)
STVT A-A	1	8.4E+00	0	uCi/cc	1.0E+00	8.4E+00	8.7E+04	0	CFM	3.4E+08

*Status: 0=Good, 1=Questionable, 2=Bad, 7=Manual Override of Raw Data

Release Characterization: Event Tree Selections:
Mix ID # 104: CONTAINMENT BYPASS RELEASE, GAP RELEASE UNCOVERED 15 - 30 MIN, NORMAL FILTRATION (0.0100)

Type	Noble Gases	Iodines	Particulates	Total	Units
Ground	3.4E+08	8.7E+05	1.9E+04	3.4E+08	uCi/sec

Start of Release: 13:53 11 / 14 / 15 End of Release: 14:53 11 / 14 / 15
Time of Trip: 13:53 11 / 14 / 15 Start of Projection 13:53 11 / 14 / 15

Dose Projection Parameters: Projection Period: 24.00 Hours Estimated Remaining Release Duration: 1.00 Hours

Dose Results

Distance	Dispersion: (s/m^3) X/Q	Projected Dose: (mrem) TEDE	Projected Dose: Child Thyroid CDE	Current Dose Rate: (mrem/hr) TEDE	Current Dose Rate: Child Thyroid CDE	Current Dose Rate: EDE	TEDE/EDE at TEDE Rate
EAB / Site Boundary	1.6E-05	1.0E+03	4.4E+03	1.4E+03	4.5E+03	1.3E+03	1.08
2 miles	4.6E-06	3.7E+02	1.2E+03	1.4E-01	3.1E-01	1.4E-01	
5 miles	1.3E-06	1.0E+02	3.1E+02	0.0E+00	0.0E+00	0.0E+00	
10 miles	4.9E-07	3.1E+01	1.1E+02	0.0E+00	0.0E+00	0.0E+00	

PAG Limit is Exceeded
10% of PAG Limit Exceeded

Dose Assessment Data Form

SITE: DB ACTUAL

MENU: B

Insufficient Data to Determine Lake Breeze

Summary For Calculation: 11/14/15 14:00

DATE PRINTED: 11/14/15 14:03

Current Input Data

Meteorological Data:	Level	Wind Direction from	Adjusted Wind Speed	Measured Wind Speed	Precipitation inch/15 min	Stability Class
	Lower	90 degrees	9.0 mph	9.0 mph	No Rain	D

Monitor ID	Release Point	Monitor Reading	Monitor Status*	Monitor Units	Conversion to uCi/cc	Effective Concentration (uCi/cc)	Flow Reading	Flow Status*	Flow Units	Release Rate (uCi/sec)
STVT A-A	1	8.4E-01	0	uCi/cc	1.0E+00	8.4E-01	8.7E+04	0	CFM	3.4E+07

*Status: 0=Good, 1=Questionable, 2=Bad, 7=Manual Override of Raw Data

Release Characterization: Event Tree Selections:

Mix ID # 104: CONTAINMENT BYPASS RELEASE, GAP RELEASE UNCOVERED 15 - 30 MIN, NORMAL FILTRATION (0.0100)

Type	Noble Gases	Iodines	Particulates	Total	Units
Ground	3.4E+07	8.7E+04	1.9E+03	3.4E+07	uCi/sec

Start of Release: 13:53 11 / 14 / 15 End of Release: 14:53 11 / 14 / 15

Time of Trip: 13:53 11 / 14 / 15 Start of Projection 13:53 11 / 14 / 15

Dose Projection Parameters: Projection Period: 24.00 Hours Estimated Remaining Release Duration: 1.00 Hours

Dose Results

Distance	Dispersion: (s/m^3) X/Q	Projected Dose: (mrem) TEDE	Projected Dose: Child Thyroid CDE	Current Dose Rate: (mrem/hr) TEDE	Current Dose Rate: Child Thyroid CDE	Current Dose Rate: EDE	TEDE/EDE at TEDE Rate
EAB / Site Boundary	1.6E-05	1.0E+02	4.4E+02	1.4E+02	4.5E+02	1.3E+02	1.08
2 miles	4.6E-06	3.7E+01	1.2E+02	2.3E-02	5.0E-02	2.2E-02	
5 miles	1.3E-06	1.0E+01	3.1E+01	0.0E+00	0.0E+00	0.0E+00	
10 miles	4.9E-07	3.1E+00	1.1E+01	0.0E+00	0.0E+00	0.0E+00	

10% of PAG Limit Exceeded

Dose Assessment Data Form

SITE: DB ACTUAL

MENU: B

Insufficient Data to Determine Lake Breeze

Summary For Calculation: 11/14/15 14:00

DATE PRINTED: 11/14/15 14:08

Current Input Data

Meteorological Data:	Level	Wind Direction from	Adjusted Wind Speed	Measured Wind Speed	Precipitation inch/15 min	Stability Class
	Lower	90 degrees	9.0 mph	9.0 mph	No Rain	D

Monitor ID	Release Point	Monitor Reading	Monitor Status*	Monitor Units	Conversion to uCi/cc	Effective Concentration (uCi/cc)	Flow Reading	Flow Status*	Flow Units	Release Rate (uCi/sec)
STVTA-A	1	8.4E-02	0	uCi/cc	1.0E+00	8.4E-02	8.7E+04	0	CFM	3.4E+06

*Status: 0=Good, 1=Questionable, 2=Bad, 7=Manual Override of Raw Data

Release Characterization: Event Tree Selections:

Mix ID # 104: CONTAINMENT BYPASS RELEASE, GAP RELEASE UNCOVERED 15 - 30 MIN, NORMAL FILTRATION (0.0100)

Type	Noble Gases	Iodines	Particulates	Total	Units
Ground	3.4E+06	8.7E+03	1.9E+02	3.4E+06	uCi/sec

Start of Release: 13:53 11 / 14 / 15 End of Release: 14:53 11 / 14 / 15

Time of Trip: 13:53 11 / 14 / 15 Start of Projection 13:53 11 / 14 / 15

Dose Projection Parameters: Projection Period: 24.00 Hours Estimated Remaining Release Duration: 1.00 Hours

Dose Results

Distance	Dispersion: (s/m^3) X/Q	Projected Dose: (mrem) TEDE	Projected Dose: (mrem) Child Thyroid CDE	Current Dose Rate: (mrem/hr) TEDE	Current Dose Rate: (mrem/hr) Child Thyroid CDE	Current Dose Rate: (mrem/hr) EDE	TEDE/EDE at TEDE Rate
EAB / Site Boundary	1.6E-05	1.0E+01	4.4E+01	1.4E+01	4.5E+01	1.3E+01	1.08
2 miles	4.6E-06	3.7E+00	1.2E+01	3.8E-03	8.2E-03	3.7E-03	
5 miles	1.3E-06	1.0E+00	3.1E+00	0.0E+00	0.0E+00	0.0E+00	
10 miles	4.9E-07	3.1E-01	1.1E+00	0.0E+00	0.0E+00	0.0E+00	

Radiation Monitor Readings for Core Uncovery During Refueling (EALs CG7 and CS7)

(REVISION 0)

Document Author: Scott McCain

Document Reviewer: Greg Van Wey

Author: Scott McCain — 10/29/07 Date

Review: Greg Van Wey — 10/31/07 Date

TABLE OF CONTENTS

1. PURPOSE

Documentation of the assumptions, calculations and results for the radiation monitor values used to indicate a loss of RPV inventory when the plant is in the cold shutdown and refueling modes for the EAL thresholds listed below.

- CG7.2.a (NEI CG1.2.a)
- CS7.3.b (NEI CS1.3)

2. DISCUSSION

The Davis Besse EAL Technical Basis Document contains the bases and references for the site specific EAL threshold values used to implement the NEI 99-01 Rev. 5 guidance methodology. This document has been developed to provide additional detailed technical documentation on specifically how the containment radiation monitor value used to indicate core uncover when shutdown was derived.

The generic guidance provided in NEI 99-01 is not intended to be used "as is." It is intended to give the logic for developing site specific EAL threshold values (Section 5.3 of NEI 99-01).

NEI 99-01 Rev. 5 provides the following guidance for CG7 and CS7 as indicated by a containment radiation monitor reading:

> *As water level in the RPV lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in site specific monitor indication and possible alarm.*
>
> *This EAL should conservatively estimate a site specific dose rate setpoint indicative of core uncovery (i.e., level at TOAF).*

A single value that would indicate core uncovery at all times cannot be set since several variables affect the source term (such as core loading, fuel burnup and time after shutdown). Therefore, a conservative set of assumptions must be established so that the calculated value will be low enough to be a valid indication of fuel uncovery for most conditions during refueling mode, but high enough so that inadvertent classifications are not made due to planned maintenance events or other conditions which may increase radiation levels.

In most PWRs, the containment high range radiation monitors are either located high in the containment dome or on the outer walls and are designed to indicate fuel damage following a Loss of Coolant Accident (LOCA). Because of their design purpose and location, these monitors are not the best choice for indication of fuel uncovery. For this reason, the refuel bridge portable radiation monitor will be used as the basis to indicate uncovery of the core during refueling conditions at Davis Besse.

3. DETERMINATION OF VALUES

3.1. Assumptions and Input Constants

3.1.1. Source Term Activity

1. Source activity is from NUREG-1228 Table 2.2 Fission Product Inventories (Ci/MWe). Although this list represents a sub-set of the total inventory, it is based on the significant dose contributors. This results in lower (conservative) estimated radiation levels.

2. Davis-Besse rated power of 950 MWe.

3. Core inventory is decay corrected for 14 days, based on a nominal refuel outage duration. This results in lower (conservative) estimated radiation levels, which will be reached quicker when the source term is higher and the ability to cool the core is more critical.

4. Only decayed activities greater than 1 Curie are used in the calculation. Limiting the isotopes in this manner will result in lower (conservative) estimated radiation levels.

5. No exposure rate is attributed to activation materials in the reactor vessel such as vessel walls and internals.

See Attachment 1 for the source term activity values.

3.1.2. Source Material Mass and Density

1. From the DB Unit 1 CY 15 Reload Report, AREVA 2006.
   - 177 fuel bundles in a fully loaded core.
   - Total bundle weight is 7.06E+05 gm (see reference 5.5)
   - 4.88E+05 gm Uranium per bundle.
   - 2.18E+05 gm Zirconium[1] per bundle.

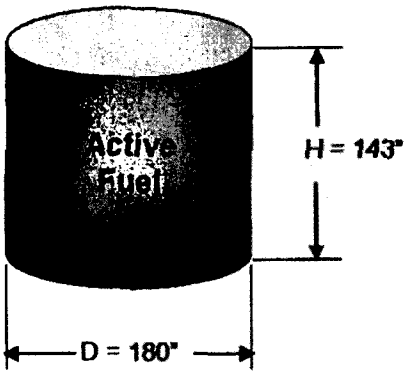
2. Source material density is taken from MicroShield (Uranium 18.75, Zirconium 6.5 and water 1.0 gm/cm^3).

3.1.3. Source Geometry

1. From the DB Unit 1 CY 15 Reload Report, AREVA 2006.
   - The active fuel length is 143".
   - The fuel assemblies (plus the gap) are 8.567" x 8.567" squares.
   - 177 assemblies in a 15'x15' circular matrix (~180" diameter).

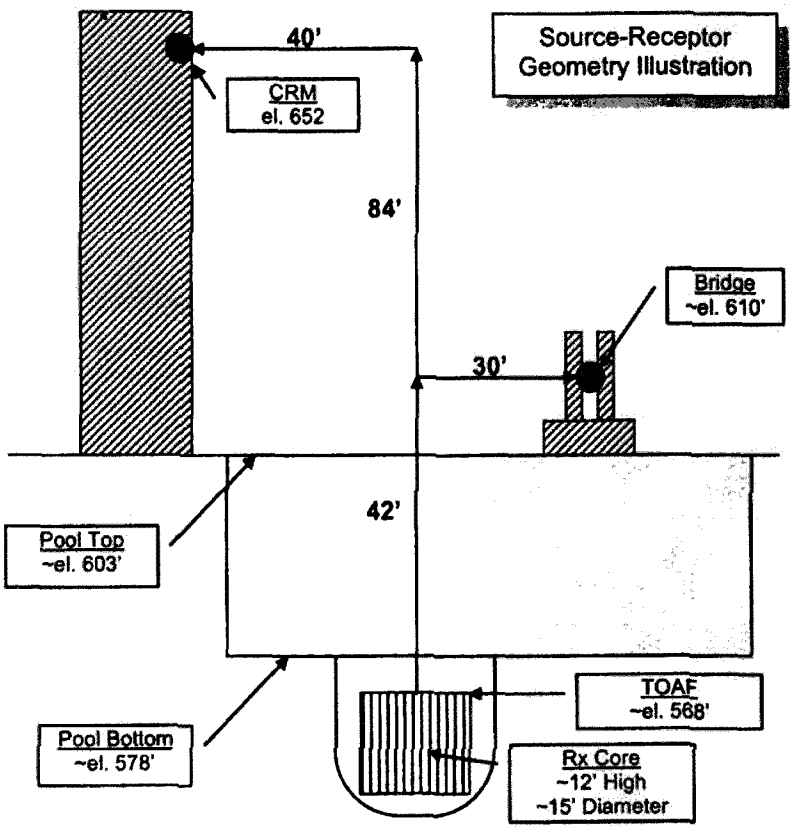
[1] The total weight of the cladding is attributed to Zirconium (versus a portion being stainless steel).

2. The source will be considered a cylinder with the core inventory evenly distributed throughout its volume.



3.1.4. Distance Between Source and Receptor Location

Two receptor points will be evaluated, the fixed containment radiation monitor and the portable refuel bridge monitor. Distances were estimated from plant drawings M-127 R5 and M-509 SH.6 R0. The value of 30' for the portable refuel bridge monitor is considered a nominal value (between directly over the core and the far edge of travel).



Note: the Y distance used in MicroShield® is the bottom of source.

3.1.5. Shielding Between Source and Receptor Location

Since the receptor points are located above the core and water level is assumed at top of active fuel, the following shields are considered in the calculations:

1. The material density of the source itself has shielding effects. See section 3.2.2 for discussion on determining source density.

2. Air – Distance from receptor points to the source (the MicroShield® program calculates the linear distance using the axial and radial values).

3. Steel (iron) – portion of fuel assembly above top of active fuel. Since this distance contains many voids (coolant channels, fuel rod springs) shield length will be set at ~10% of actual distance. For the Davis-Besse fuel assemblies, the distance from TOAF to top of the fuel assembly is ~ 14", therefore the effective shield distance is set a 1.4".

3.2. Calculations

3.2.1. Source Volume

1. Source Cylinder Total Volume:

$$V = \pi r^2 h \times 16.387$$

$$5.96E{+}7\ cm^3 = \pi \times (90\ in)^2 \times 143\ in \times 16.387\ cm^3/in^3$$

2. Source Materials Volume:

   See Attachment 2 for the source materials volume calculations.

3.2.2. Source Materials Mixed Shielding Densities

1. Uranium and Zirconium volume is derived as follows:

$$Volume_{cm^3} = \frac{Weight_{gm\ per\ bundle} \times \#\ bundles}{Density_{gm/cm^3}}$$

2. Water volume is derived as follows:

$$Water\ Vol = Source\ Vol - (U\ Vol + Zirc\ Vol)$$

3. Material Volume Fractions (MVF) per MicroShield® instructions are derived as follows:

$$MVF = \frac{Material\ Volume_{cm^3}}{Source\ Volume_{cm^3}}$$

4. Individual Density Entries (IDE) per MicroShield® instructions are derived as follows:

$$IDE_{gm/cc} = MVF \times Material\ Density_{gm/cc}$$

See Attachment 2 for the source materials mixed shielding densities calculations.

4. CONCLUSIONS

Containment Radiation Monitor (R/hr)	Refuel Bridge Portable Monitor (R/hr)
1.58E+01	2.97E+01

5. REFERENCES

5.1. NUREG-1228, October 1988, Table 2.2, Fission Product Inventories

5.2. Nuclear Decay Data for Radionuclides Occuring in Routine Releases from Nuclear Fuel Cycle, D.C.Kocher

5.3. Drawing M-127 R5, Equipment Locations Containment Vessel Section "A" – "A"

5.4. Drawing M-509 SH.6 R0, Equipment Elevations Inside Containment

5.5. DB CY 15 Reload Report, AREVA 2006

5.6. 51-5016988-00, DB CY 14 Design Report Table 5-2

6. ATTACHMENTS

6.1. Source Term Decay Corrected Core Inventory

6.2. MicroShield® Source Geometry and Density Inputs

6.3. MicroShield® Report

	Decay Constant (-hrs)	NUREG-1228 Table 2.2 (Ci/Mwe)	NUREG-1228 DB Core Inventory (Ci)	Decay Corrected Core Inventory (Ci)	Selected
Kr-85	7.38E-06	5.60E+02	5.32E+05	5.31E+05	✓
Kr-85m	1.55E-01	2.40E+04	2.28E+07	5.49E-16	
Kr-87	5.45E-01	4.70E+04	4.47E+07	1.32E-72	
Kr-88	2.44E-01	6.80E+04	6.46E+07	1.60E-28	
Xe-131m	2.42E-03	1.00E+03	9.50E+05	4.21E+05	✓
Xe-133	5.51E-03	1.70E+05	1.62E+08	2.54E+07	✓
Xe-133m	1.32E-02	6.00E+03	5.70E+06	6.76E+04	✓
Xe-135	7.61E-02	3.40E+04	3.23E+07	2.54E-04	
Xe-138	2.94E+00	1.70E+05	1.62E+08	0.00E+00	
I-131	3.60E-03	8.50E+04	8.08E+07	2.41E+07	✓
I-132	3.01E-01	1.20E+05	1.14E+08	1.36E-36	
I-133	3.33E-02	1.70E+05	1.62E+08	2.23E+03	✓
I-134	7.91E-01	1.90E+05	1.81E+08	6.78E-108	
I-135	1.05E-01	1.50E+05	1.43E+08	6.79E-08	
Cs-134	3.84E-05	7.50E+03	7.13E+06	7.03E+06	✓
Cs-136	2.21E-03	3.00E+03	2.85E+06	1.36E+06	✓
Cs-137	2.63E-06	4.70E+03	4.47E+06	4.46E+06	✓
Te-129m	8.60E-04	5.30E+03	5.04E+06	3.77E+06	✓
Te-131m	2.14E-02	1.30E+04	1.24E+07	9.34E+03	✓
Te-132	8.86E-03	1.20E+05	1.14E+08	5.80E+06	✓
Sb-127	7.52E-03	6.10E+03	5.80E+06	4.64E+05	✓
Sb-129	1.58E-01	3.30E+04	3.14E+07	3.26E-16	
Sr-89	5.73E-04	9.40E+04	8.93E+07	7.37E+07	✓
Sr-90	2.72E-06	3.70E+03	3.52E+06	3.51E+06	✓
Sr-91	7.29E-02	1.10E+05	1.05E+08	2.37E-03	
Ba-140	2.26E-03	1.60E+05	1.52E+08	7.10E+07	✓
Ru-103	7.35E-04	1.10E+05	1.05E+08	8.16E+07	✓
Ru-106	7.84E-05	2.50E+04	2.38E+07	2.31E+07	✓
Mo-99	1.05E-02	1.60E+05	1.52E+08	4.44E+06	✓
La-140	1.72E-02	1.60E+05	1.52E+08	4.70E+05	✓
Y-91	4.95E-04	1.20E+05	1.14E+08	9.65E+07	✓
Ce-144	1.02E-04	8.50E+04	8.08E+07	7.80E+07	✓
Np-239	1.23E-02	1.64E+06	1.56E+09	2.53E+07	✓

DB Rated Power (MWe): 950
TAS (days): 14

MicroShield Source Materials Instructions

Materials may be "mixed" for a shield by entering partial densities of more than one material. A shield may be composed of built-in materials and custom materials with either specified by its density. A density of zero for any non-zero shield is not allowed.

When more than one material is specified for a shield, the user has two choices to decide the appropriate density. One choice is to pre-calculate the volume fraction or the effective density for each material in the region and use that as an entry. As an example of pre-calculated volume fractions, a source consists of a cylindrical tank of water with steel internal structures. The source region may be represented as a tank with a homogeneous mixture of steel and water. The volume of the tank is 10,000 liters and it contains 8,000 kg of water and 1560 kg of iron. The material densities to enter are then:

Water: = *0.8 gm/cc*

Iron: = *0.156 gm/cc*

Another way to estimate densities is to multiply each individual material volume fraction by its pure density. For example, if a volume contains 95% (by volume) water at a density of 1.0 gm/cc and 5% (by volume) steel at a density of 7.85 gm/cc, then the individual density entries are:

Water: *0.95 x 1.0 gm/cc = 0.95 gm/cc*

Iron: *0.05 x 7.85 gm/cc = 0.3925 gm/cc*

Source Height (in):	143
Source Radius (in):	90
Source Volume (cm3):	5.96E+07

# Fuel Bundles:	177
Bundle Weight (gm):	7.06E+05
Uranium Weight (gm per bundle):	4.88E+05
Cladding Weight (gm per bundle):	2.18E+05

	Density	Volume (cm3)	Material Volume Fraction - MVF	Individual Density Entries - IDE (gm/cc)
Uranium	18.75	4.61E+06	7.73%	1.45
Zirconium	6.5	5.94E+06	9.96%	0.65
Water	1	4.91E+07	82.32%	0.82

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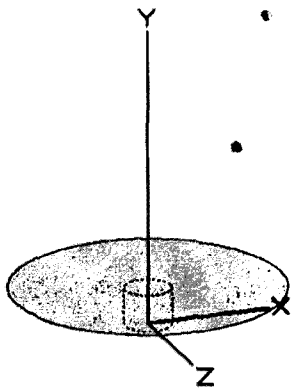
Date: _____
By: _____
Checked: _____

Case Title: Davis-Besse CG7/CS7
Description: Core Uncovery Calculations - 14 Day Decay
Geometry: 8 - Cylinder Volume - End Shields



Source Dimensions

Height	363.22 cm	11 ft 11.0 in
Radius	228.6 cm	7 ft 6.0 in

Dose Points

	X	Y	Z
# 1	914.4 cm	1645.92 cm	0 cm
	30 ft	54 ft 0.0 in	0.0 in
# 2	1219.2 cm	2926.08 cm	0 cm
	40 ft 0.0 in	96 ft	0.0 in

Shields

Shield Name	Dimension	Material	Density
Source	5.96e+07 cm³	Mixed ->	2.92
		Uranium	1.45
		Water	0.82
		Zirconium	0.65
Shield 1	3.556 cm	Iron	7.86
Air Gap		Air	0.00122
Immersion		Air	0.00122

Source Input
Grouping Method : Standard Indices
Number of Groups : 25
Lower Energy Cutoff : 0.015
Photons < 0.015 : Excluded
Library : Grove

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Ba-140	7.1000e+007	2.6270e+018	1.1907e+006	4.4054e+010
Ce-144	7.8000e+007	2.8860e+018	1.3080e+006	4.8398e+010
Cs-134	7.0300e+006	2.6011e+017	1.1789e+005	4.3620e+009
Cs-136	1.3600e+006	5.0320e+016	2.2807e+004	8.4386e+008
Cs-137	4.4600e+006	1.6502e+017	7.4793e+004	2.7674e+009
I-131	2.4100e+007	8.9170e+017	4.0415e+005	1.4954e+010
I-133	2.2300e+003	8.2510e+013	3.7397e+001	1.3837e+006
Kr-85	5.3100e+005	1.9647e+016	8.9048e+003	3.2948e+008
La-140	4.7000e+005	1.7390e+016	7.8818e+003	2.9163e+008
Mo-99	4.4400e+006	1.6428e+017	7.4458e+004	2.7549e+009
Np-239	2.5300e+007	9.3610e+017	4.2428e+005	1.5698e+010
Ru-103	8.1600e+007	3.0192e+018	1.3684e+006	5.0631e+010
Ru-106	2.3100e+007	8.5470e+017	3.8738e+005	1.4333e+010
Sb-127	4.6400e+005	1.7168e+016	7.7812e+003	2.8790e+008

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Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Sr-89	7.3700e+007	2.7269e+018	1.2359e+006	4.5730e+010
Sr-90	3.5100e+006	1.2987e+017	5.8862e+004	2.1779e+009
Te-129m	3.7700e+006	1.3949e+017	6.3222e+004	2.3392e+009
Te-131m	9.3400e+003	3.4558e+014	1.5663e+002	5.7953e+006
Te-132	5.8000e+006	2.1460e+017	9.7265e+004	3.5988e+009
Xe-131m	4.2100e+005	1.5577e+016	7.0601e+003	2.6122e+008
Xe-133	2.5400e+007	9.3980e+017	4.2595e+005	1.5760e+010
Xe-133m	6.7600e+004	2.5012e+015	1.1336e+003	4.1945e+007
Y-91	9.6500e+007	3.5705e+018	1.6183e+006	5.9877e+010

Buildup

The material reference is : Source

This buildup reference material is a mixed material with a high atomic number element (92). Buildup Factors less than and somewhat greater than 116 keV may be incorrect. Please understand your results.

Integration Parameters

Radial	20
Circumferential	10
Y Direction (axial)	10

Results - Dose Point # 1 - (914.4,1645.92,0) cm

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.02	3.166e+16	0.000e+00	3.611e-19	0.000e+00	1.251e-20
0.03	1.106e+18	9.873e-159	7.985e-17	9.784e-161	7.913e-19
0.04	2.712e+17	3.580e-69	1.727e-16	1.583e-71	7.639e-19
0.05	4.035e+16	4.314e-36	1.698e-16	1.149e-38	4.523e-19
0.06	2.073e+16	8.207e-21	4.528e-16	1.630e-23	8.994e-19
0.08	4.176e+17	1.641e-06	1.504e-03	2.597e-09	2.380e-06
0.1	6.863e+17	2.327e-01	1.046e+02	3.561e-04	1.600e-01
0.15	5.109e+17	9.749e+00	2.880e+03	1.605e-02	4.742e+00
0.2	3.445e+17	1.483e+03	1.324e+05	2.617e+00	2.337e+02
0.3	3.950e+17	8.939e+04	3.128e+06	1.696e+02	5.934e+03
0.4	8.754e+17	1.322e+06	2.839e+07	2.575e+03	5.532e+04
0.5	3.382e+18	1.606e+07	2.354e+08	3.153e+04	4.620e+05
0.6	5.895e+17	6.080e+06	6.693e+07	1.187e+04	1.306e+05
0.8	3.505e+17	1.011e+07	7.564e+07	1.923e+04	1.439e+05
1.0	7.053e+16	4.074e+06	2.364e+07	7.509e+03	4.357e+04
1.5	2.458e+16	4.237e+06	1.699e+07	7.129e+03	2.859e+04
2.0	1.621e+14	5.323e+04	1.757e+05	8.231e+01	2.717e+02
3.0	6.192e+14	4.378e+05	1.161e+06	5.940e+02	1.575e+03

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Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	9.117e+18	4.247e+07	4.516e+08	8.069e+04	8.721e+05

Results - Dose Point # 2 - (1219.2,2926.08,0) cm

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.02	3.166e+16	0.000e+00	1.191e-19	0.000e+00	4.125e-21
0.03	1.106e+18	4.782e-170	2.633e-17	4.740e-172	2.609e-19
0.04	2.712e+17	4.973e-75	5.695e-17	2.199e-77	2.519e-19
0.05	4.035e+16	1.110e-39	5.599e-17	2.956e-42	1.491e-19
0.06	2.073e+16	2.759e-23	1.461e-16	5.480e-26	2.901e-19
0.08	4.176e+17	5.258e-08	5.819e-05	8.320e-11	9.209e-08
0.1	6.863e+17	1.846e-02	9.984e+00	2.824e-05	1.527e-02
0.15	5.109e+17	4.476e-01	1.834e+02	7.370e-04	3.020e-01
0.2	3.445e+17	1.713e+02	2.043e+04	3.023e-01	3.606e+01
0.3	3.950e+17	2.382e+04	1.026e+06	4.519e+01	1.946e+03
0.4	8.754e+17	4.438e+05	9.976e+06	8.647e+02	1.944e+04
0.5	3.382e+18	5.593e+06	8.122e+07	1.098e+04	1.594e+05
0.6	5.895e+17	2.117e+06	2.270e+07	4.131e+03	4.431e+04
0.8	3.505e+17	3.471e+06	2.518e+07	6.602e+03	4.789e+04
1.0	7.053e+16	1.382e+06	7.799e+06	2.547e+03	1.438e+04
1.5	2.458e+16	1.414e+06	5.561e+06	2.379e+03	9.356e+03
2.0	1.621e+14	1.765e+04	5.737e+04	2.730e+01	8.872e+01
3.0	6.192e+14	1.446e+05	3.788e+05	1.962e+02	5.139e+02
TOTALS:	9.117e+18	1.461e+07	1.539e+08	2.777e+04	2.974e+05