ENCLOSURE 1
RBG-47847

EVALUATION OF PROPOSED CHANGES

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1.0 SUMMARY DESCRIPTION

Entergy Operations, Inc. (Entergy) proposes to revise the River Bend Station, Unit 1(RBS) currently approved Emergency Plan (EP) Emergency Action Level (EAL) scheme, which is based on the Nuclear Energy Institute's (NEI's) guidance established in NEI 99-01, Revision 5, *"Methodology for Development of Emergency Action Levels"* (Reference 1). Entergy is proposing to adopt the EAL schemes based on the guidance provided in NEI 99-01, Revision 6, *"Development of Emergency Action Levels for Non-Passive Reactors,"* which has been endorsed by the NRC (Reference 2)

2.0 DETAILED DESCRIPTION

The proposed changes involve revising RBS's EAL scheme, which is currently based on NEI 99-01, Revision 5, to a scheme based on NEI 99-01, Revision 6. Enhancements provided by Revision 6 of the guidance include:

- Clarification of numerous EALs that have been typically misinterpreted by the industry in the development of their site-specific EAL scheme,
- Clarification of the intent of EALs that have been historically misclassified,
- Providing additional guidance for the development of EALs for current non-passive reactor designs as well as possible future reactor designs that are non-passive,
- Incorporating lessons learned from industry events (i.e., Fukushima and others) and NUREG/CR-7154, *"Risk Informing Emergency Preparedness Oversight: Evaluation of Emergency Action Levels – A Pilot Study of Peach Bottom, Surry and Sequoyah,"* and
- A detailed review of the guidance to re-validate that the EALs are appropriate and are at the necessary emergency classification level based upon 32 years of industry and NRC experience with EAL scheme development and implementation.

2.1 Proposed Initiating Conditions (ICs) and EALs

Enclosure 2, *"Proposed EAL Technical Basis Document (Markup),"* provides a markup of the current RBS EAL scheme basis illustrating changes incorporating the guidance of NEI 99-01, Revision 6. Enclosure 3 provides a clean copy of RBS EAL Technical Basis document.

2.2 Deviations and Differences

Enclosure 4 contains a matrix that provides a comparison of the Initiating Conditions (ICs) and EALs in NEI 99-01 to the ICs and EALs proposed for RBS. The comparison evaluates differences and deviations consistent with the similar exercise performed during the RBS EP upgrade to NEI 99-01, Revision 5 (Reference 3). The NRC approved RBS's EP upgrade to NEI 99-01, Revision 5, on July 31, 2012 (Reference 4).

A *difference* is an EAL change where 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 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 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.).

A number of differences are the result of adding plant-specific information to the EALs. In these cases, Enclosure 4 may refer the reader to an associated document in Enclosure 5, *"Supporting Referenced Document Pages,"* which provides the technical basis for plant-specific information.

2.3 Generic Differences

The differences below apply throughout the set of EALs and are not specifically identified in each instance in the comparison matrix as a difference.

NEI 99-01, Rev 6 EALs		**RBS Station EALs**
References PWRs		Deleted PWR references as appropriate
Uses E-HU for ISFSI ICs		Uses EU for ISFSI ICs
Designates ICs and EALs as Example: (IC)HU1 EAL 2		Designates ICs and EALs as Example: HU1.2
Emergency Classification ICs are presented together by Emergency Classification level (all NOUEs grouped together, then all Alerts, etc.) for each category (A, C, H, etc.), in ascending order (UE – GE)		Emergency Classification ICs are presented by Emergency Classification "family" (A…1, A…2, A…3, etc.) for each category, in ascending order (UE – GE)

2.4 Operational Modes

Mode applicability of the proposed ICs and EALs is consistent with the NEI 99-01, Revision 6. The following tables provide the operating modes for RBS as defined by the respective Technical Specifications (TSs).

RBS

MODE	TITLE	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	NA
2	Startup	NA
3	Hot Shutdown [(a)]	> 200
4	Cold Shutdown [(a)]	≤ 200
5	Refueling [(b)]	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.[CA1]

In addition to the TS defined operational modes, NEI 99-01, Revision 6, defines the following additional mode:

Defueled: All reactor fuel removed from the reactor vessel (i.e., full core off load during refueling or extended outage).

RBS procedures recognize and are consistent with the definition of a defueled condition.

2.5 <u>Instrumentation Used for EALs</u>

RBS has verified that the specified values used as EAL setpoints are within the calibrated range of the referenced instrumentation.

2.6 <u>Background Technical Information</u>

Enclosure 2 provides the existing EAL Technical Basis document marked to illustrate the proposed changes. Enclosure 3 provides the revised (clean) EAL Technical Basis document. Enclosure 4 provides a deviation-difference document comparing NEI 99-01, Revision 6, with the proposed changes to the RBS EAL schemes. Enclosure 5 provides specific reference documents or excerpts which support related RBS Emergency Plan proposed changes.

3.0 TECHNICAL EVALUATION

Entergy has evaluated the proposed changes to determine whether applicable regulations and requirements have been met. NEI 99-01 guidance methodology includes many years of development, along with use and implementation. The guidance has been subject to NRC reviews and approval. The RBS EAL scheme currently in place is based on the methodology in NEI 99-01, Revision 5. NEI 99-01, Revision 6, is the latest version endorsed by the NRC and provides guidance to nuclear power plant operators for the development of a site-specific emergency action level scheme.

10 CFR 50.47(b)(4) requires that emergency plans include a standard emergency classification and action level scheme. This scheme is a fundamental component of an EP in that it provides the defined thresholds that will allow site personnel to rapidly implement a range of pre-planned emergency response measures. An emergency classification scheme also facilitates timely decision-making by an offsite response organization concerning the implementation of precautionary or protective actions for the public.

NEI 99-01, Revision 6, contains a generic set of ICs, EALs, and fission product barrier status thresholds. The guidance also includes supporting technical basis information, developer notes, and recommended classification instructions for users. The methodology described in this document is consistent with NRC requirements and guidance. In particular, this methodology was specifically endorsed by the NRC in a March 28, 2013, letter from NRC to NEI (Reference 2) and determined to provide an acceptable approach in meeting requirements of 10 CFR 50.47(b)(4), applicable requirements of 10 CFR 50, Appendix E, and the associated planning standard evaluation elements contained in NUREG-0654/FEMA-REP-1, Revision 1, *"Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,"* November 1980.

10 CFR 50, Appendix E, Section IV.B.2, requires 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 change to the RBS EAL scheme from NEI 99-01, Revision 5, to NEI 99-01, Revision 6, guidance does not reduce the capability to meet the applicable emergency planning standards and requirements in 10 CFR 50.47(b) and 10 CFR 50, Appendix E. Accordingly, pursuant to the requirements of 10 CFR 50, Appendix E, Section IV.B.2, Entergy requests NRC review and approval of the proposed changes to the EAL scheme as a license amendment request in accordance with 10 CFR 50.90.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The regulations in 10 CFR 50.54(q) provide direction to licensees seeking to revise emergency plans. The requirements related to nuclear power plant emergency plans are contained in the standards in 10 CFR 50.47, "Emergency Plans," and the requirements of Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities."

Paragraph 10 CFR 50.47(a)(1) states that no operating license for a nuclear power reactor will be issued unless a finding is made by the NRC that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Section 50.47(b) contains standards that onsite and offsite emergency response plans must meet for the NRC staff to make a positive finding that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. One of these standards, 10 CFR 50.47(b)(4), requires that emergency plans include a standard emergency classification and action level scheme.

10 CFR 50, Appendix E, Section IV.B, *"Assessment Actions,"* requires that emergency plans include emergency action levels (EALs) that are to be used as criteria for determining the need for notification and participation of local and state agencies, and for determining when and what type of protective measures should be considered to protect the health and safety of individuals both onsite and offsite. EALs are to be based on plant conditions and instrumentation, as well as onsite and offsite radiological monitoring. Section IV.B provides that initial EALs shall be discussed and agreed on by the applicant and state and local authorities, be approved by the NRC, and reviewed annually thereafter with state and local authorities. Therefore, a revision to EALs will require NRC approval prior to implementation, if it involves (1) changing from one EAL scheme to another (e.g., NEI 99-01, Revision 4 to NEI 99-01, Revision 6), (2) proposing an alternate method to comply with the regulations, or (3) the EAL revision proposed by the licensee decreases the effectiveness of the emergency plan.

NRC Regulatory Issue Summary (RIS) 2005-02, Revision 1, "Clarifying the Process for Making Emergency Plan Changes", issued April 19, 2011, says that a change in an EAL scheme to incorporate the improvements provided in NUMARC/NESP-007 or NEI 99-01 would not decrease the overall effectiveness of the emergency plan, but due to the potential safety significance of the change, the change needs prior NRC review and approval.

The proposed changes meet the above regulatory requirements.

4.2 Precedent

The following commercial nuclear power plants have received license amendments that approved EALs based on NEI 99-01, Revision 6:

Callaway - Amendment 212 (Reference 5)

Fermi 2 - Amendment 202 (Reference 6)

South Texas - Amendment 206 and 194 (Reference 7)

4.3 No Significant Hazards Consideration Analysis

Entergy Operations, Inc. (Entergy) proposes to revise the currently approved Emergency Plan (EP) Emergency Action Level (EAL) scheme for River Bend Station, Unit 1 (RBS), which is based on the Nuclear Energy Institute's (NEI's) guidance established in NEI 99-01, Revision 5, *"Methodology for Development of Emergency Action Levels."* Entergy is proposing to adopt the EAL schemes based on the guidance provided in NEI 99-01, Revision 6, *"Development of Emergency Action Levels for Non-Passive Reactors,"* which has been endorsed by the NRC.

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, *"Issuance of Amendment."* As required by 10 CFR 50.91(a), Entergy analysis of the issue of no significant hazards consideration is presented below.

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

   Response: No

   The proposed changes to the RBS EALs do not involve any physical changes to plant equipment or systems and do not alter the assumptions of any accident analyses. The proposed changes do not adversely affect accident initiators or precursors and do not alter design assumptions, plant configuration, or the manner in which the plant is operated and maintained. The proposed changes do not adversely affect the ability of structures, systems or components (SSCs) to perform intended safety functions in mitigating the consequences of an initiating event within the assumed acceptance limits.

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

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

   Response: No

   No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The changes do not challenge the integrity or performance of any safety-related systems. No plant equipment is installed or removed, and the changes do not alter the design, physical configuration, or method of operation of any plant SSC. Because EALs are not accident initiators and no physical changes are made to the plant, no new causal mechanisms are introduced.

   Therefore, the changes do not create the possibility of a new or different kind of accident from an accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No

   Margin of safety is associated with the ability of the fission product barriers (i.e., fuel cladding, reactor coolant system pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed changes do not impact operation of the plant and no accident analyses are affected by the proposed changes. The changes do not affect the Technical Specifications or the method of operating the plant. Additionally, the proposed changes will not relax any criteria used to establish safety limits and will not relax any safety system settings. The safety analysis acceptance criteria are not affected by these changes. The proposed changes will not result in plant operation in a configuration outside the design basis. The proposed changes do not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition.

   Therefore, the changes do not involve a significant reduction in a margin of safety.

Based upon the reasoning presented above, Entergy concludes that the requested change involves no significant hazards consideration, as set forth in 10 CFR 50.92(c), *"Issuance of Amendment."*

4.4 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 are applicable to emergency planning requirements involving the proposed adoption of the NRC-endorsed EAL guidance as described in NEI 99-01, Revision 6, and do not reduce the capability to meet the emergency planning standards of 10 CFR 50.47(b) and the requirements of 10 CFR 50, Appendix E. The proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed changes.

6.0 REFERENCES

1. NEI 99-01, Revision 5, *"Methodology for Development of Emergency Action Levels"* February 2008 (ML080450149)

2. NRC letter *"U.S. Nuclear Regulatory Commission Review and Endorsement of NEI 99-01, Revision 6, Dated November, 2012 (TAC No. D92368),"* March 28, 2013 (ML12346A463)

3. Entergy letter dated August 1, 2011, *"Proposed Emergency Action Levels Using NEI 99-01 Revision 5 Scheme,"* (ML11216A055) (RBG-47165)

4. NRC Safety Evaluation dated July 31, 2012, *"River Bend Station, Unit 1 - Emergency Action Level Scheme Upgrade Based on Nuclear Energy Institute (NEI) 99-01, Revision 5, "Methodology for Development of Emergency Action Levels"* (TAC Nos. ME6846) (ML12178A567) (RBC-51043)

5. NRC letter *"Callaway Plant, Unit 1 - Issuance of Amendment Re: Upgrade to Emergency Action Level Scheme (CAC No. MF4945),"* October 7, 2015 (ML15251A493)

6. NRC letter *"Fermi 2 - Issuance of Amendment to Revise the Emergency Action Level Scheme for the FERMI 2 Emergency Plan (TAC No. MF5048),"* September 29, 2015 (ML15233A084)

7. NRC letter *"South Texas Project, Units 1 and 2 - Re: Upgrade to Emergency Action Level Scheme (TAC Nos. MF4195 and MF4196),"* August 20, 2015 (ML15201A195)

ENCLOSURE 2
RBG-47847

PROPOSED EAL TECHNICAL BASIS DOCUMENT (NEI REVISION 6 MARKUP)

River Bend Station
EAL Technical Basis

Table of Contents

1.0 INTRODUCTION

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for River Bend Station (RBS). It should be used to facilitate review of the RBS EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EIP-2-001, Classification of Emergencies, 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 to be made as soon as conditions are present and recognizable for the classification, but within 15 minutes or less in all cases of conditions present. Use of this document for assistance is not intended to delay the emergency classification.

Because the information in a basis document can affect emergency classification decision-making (e.g., the Emergency Director 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 RBS Emergency Plan.

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

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

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

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

2.2 Fission Product Barriers

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

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

The primary fission product barriers are:

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

B. Reactor Coolant System Barrier (RCB): The RCS Barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping out to and including the isolation valves.

C. Containment Barrier (CNB): The Primary Containment Barrier includes the drywell, the containment, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. Containment Barrier thresholds are used as criteria for escalation of the Emergency Classification Level (ECL) from Alert to a Site Area Emergency or a General Emergency.

2.3 Fission Product Barrier Classification Criteria

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

Alert:

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

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

2.4 EAL Organization

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

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

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

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

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

EAL Groups, Categories and Subcategories

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

2.5 Technical Bases Information

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

<u>Category Letter & Title</u>

<u>Subcategory Number & Title</u>

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

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

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

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

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

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

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

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

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

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 Shutdown, 4 - Cold Shutdown, 5 - Refueling, DEF - Defueled, or All. (See Section 2.6 for operating mode definitions)

Definitions:

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

Basis:

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

Reference(s):

Source documentation from which the EAL is derived

2.6 Operating Mode Applicability

1 Power Operation

Reactor is critical and the mode switch is in RUN

2 Startup

The mode switch is in REFUEL (with all reactor vessel head closure bolts fully tensioned) or STARTUP/HOT STANDBY

3 Hot Shutdown

The mode switch is in SHUTDOWN and average reactor coolant temperature is >200°F

4 Cold Shutdown

The mode switch is in SHUTDOWN and average reactor coolant temperature is ≤ 200°F

5 Refueling

The mode switch is in REFUEL or SHUTDOWN with one or more reactor vessel head closure bolts are less than fully tensioned

DEF Defueled

RPV contains no irradiated fuel

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

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the 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 informing basis information. In the Recognition Category F matrices, EALs are based on loss or potential loss of Fission Product Barrier Thresholds.

EAL matrices should be read from left to right, from General Emergency to Unusual Event, and top to bottom. Declaration decisions should be independently verified before declaration is made except when gaining this verification would exceed the 15 minute declaration requirement. Place keeping should be used on all EAL matrices.

3.1.1 Classification Timeliness

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

3.1.2 Valid Indications

All emergency classification assessments shall be based upon valid indications, reports or conditions. A valid indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy. 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 to be valid when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

3.1.3 Imminent Conditions

For ICs and EALs that have a stipulated time duration (e.g., 15 minutes, 30 minutes, etc.), the Emergency 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).

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.

3.2 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. 4.1.8).

3.2.1 Classification of Multiple Events and Conditions

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

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

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

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

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

3.2.2 Consideration of Mode Changes During Classification

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

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

3.2.3 Classification of Imminent Conditions

Although EALs provide specific thresholds, the 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 Termination

An ECL may be terminated when the event or condition that meets the classified IC and EAL no longer exists, and other site-specific termination requirements are met.

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

3.2.5 Classification of Short-Lived Events

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

3.2.6 Classification of Transient Conditions

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

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

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

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

It is important to stress that the 15-minute emergency classification assessment period (process clock) is not a "grace period" during which a classification may be delayed to allow the performance of a corrective action that would obviate the need to classify the event. Emergency classification assessments must be deliberate and timely, with no undue delays. The provision discussed above addresses only those rapidly evolving situations when an operator is able to take a successful corrective action prior to the Emergency 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 ML12326A805

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

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

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

4.1.5 10 § CFR 50.73 License Event Report System

4.1.6 RBS Technical Specifications Table 1.1-1, Modes

4.1.7 RBS USAR Section 2.1 Geography and Demography

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

4.1.9 RBS Emergency Plan

4.1.10 RBS USAR 9.1.2.5 Holtec HI-STORM Dry Spent Fuel Storage System

4.1.11 RBS USAR 9.1.4.2.3.11 Fuel Transfer System

4.1.12 OSP-0037 Shutdown Operations Protection Plan (SOPP)

4.1.13 RBS Security Plan

4.2 Implementing

4.2.1 EIP-2-001 Classification of Emergencies

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

4.2.3 RBS EAL Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

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

Alert

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

Confinement Boundary

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the RBS ISFSI, Confinement Boundary is defined as the Holtec System Multi-Purpose Canister (MPC) (ref. 4.1.10).

Containment Closure

The ~~procedurally defined conditions or~~ actions taken to secure primary containment ~~(primary or secondary for BWR)~~ and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Containment Closure is established when the containment requirements of OSP-0037 (ref. 4.1.12) are met and at least one integral barrier to the release of radioactive material is provided. within the specified limits using STP-057-3804.

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 (~~N~~OUE)
- Alert
- Site Area Emergency (SAE)
- General Emergency (GE)

Explosion

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

~~Faulted~~

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

Fire

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

Fission Product Barrier Threshold

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

Flooding

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

General Emergency

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

Hostage

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

Hostile Action

An act toward ~~a NPP~~ RBS or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should **not** be construed to include acts of civil disobedience or felonious acts that are **not** part of a concerted attack on ~~the NPP~~RBS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

Hostile Force

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

Imminent

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

Impede(d)

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

Independent Spent Fuel Storage Installation (ISFSI)

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

Initiating Condition (IC)

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

~~**Normal Levels**~~

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

Owner Controlled Area (OCA)

For the purposes of classification, the Security Owner Controlled Area (SOCA) or the area between the SOCA Fence and the PROTECTED AREA Boundary (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

The area within the perimeter of the RBS security fence. (ref. 4.1.9).

Refueling Pathway

Reactor cavity (well), containment spent fuel pool, fuel transfer canal, and fuel building spent fuel pools, but **not** including the reactor vessel, comprise the refueling pathway (ref. 4.1.11).

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.~~

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

For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline. (ref. 4.1.7)

Unisolable

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

Unplanned

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

Unusual Event

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

Valid

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

Visible Damage

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

5.2 Abbreviations/Acronyms

°F .... Degrees Fahrenheit
° .... Degrees
AB .... Auxiliary Building
AC .... Alternating Current
AOP .... Abnormal Operating Procedure
APRM .... Average Power Range Meter
ATWS .... Anticipated Transient Without Scram
BWR .... Boiling Water Reactor
BWROG .... Boiling Water Reactor Owners Group
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
CNB .... Containment Barrier
CS .... Core Spray
CTMT .... Containment
DEF .... Defueled
DBA .... Design Basis Accident
DC .... Direct Current
D/G .... Diesel Generator
DRMS .... Digital Radiation Monitoring System
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
EPG .... Emergency Procedure Guideline
EPP .... Emergency Plan Procedure
ERO .... Emergency Response Organization
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
HCTL .... Heat Capacity Temperature Limit

HPCS...High Pressure Core Spray
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
LPCS ....Low Pressure Core Spray
LRW...Liquid Radwaste
LWR..Light Water Reactor
MPC.................................. Maximum Permissible Concentration/Multi-Purpose Canister
MPH...Miles Per Hour
mR, mRem, mrem, mREM ....milli-Roentgen Equivalent Man
MSCRWL...Minimum Steam Colling RPV Water Level
MSIV...Main Steam Isolation Valve
MSL ....Main Steam Line
MW ....Megawatt
NEI...Nuclear Energy Institute
NEIC ....National Earthquake Information Center
NESP..National Environmental Studies Project
NORAD...North American Aerospace Defense Command
(NO)UE...Notification of Unusual Event
NPP ....Nuclear Power Plant
NRC ....Nuclear Regulatory Commission
NSSS ....Nuclear Steam Supply System
OBE ....Operating Basis Earthquake
ODCM..Offsite Dose Calculation Manual
ORO ....Offsite Response Organization
PA...Protected Area
PAG ....Protective Action Guideline
PB...Pushbutton
PCIS ....Primary Containment Isolation System
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PSID ....Pounds Per Square Inch Differential
PSIG ....Pounds per Square Inch Gauge

R .... Roentgen
RCB .... RCS Barrier
RCIC .... Reactor Core Isolation Cooling
RCS .... Reactor Coolant System
Rem, rem, REM .... Roentgen Equivalent Man
RETS .... Radiological Effluent Technical Specifications
RHR .... Residual Heat Removal
RPS .... Reactor Protection System
RPV .... Reactor Pressure Vessel
RWCU .... Reactor Water Cleanup
SAG .... Severe Accident Guideline
SAP .... Severe Accident Procedure
SAR .... Safety Analysis Report
SBO .... Station Blackout
SCBA .... Self-Contained Breathing Apparatus
SOCA .... Security Owner Controlled Area
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
SRV .... Safety Relief Valve
SSE .... Safe Shutdown Earthquake
TEDE .... Total Effective Dose Equivalent
TAF .... Top of Active Fuel
TSC .... Technical Support Center
UFSAR .... Updated Final Safety Analysis Report
USGS .... United States Geological Survey

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

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

RBS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
AU1.1	AU1	1, 2
AU1.2	AU1	3
AU2.1	AU2	1
AA1.1	AA1	1
AA1.2	AA1	2
AA1.3	AA1	3
AA1.4	AA1	4
AA2.1	AA2	1
AA2.2	AA2	2
AA2.3	AA2	3
AA3.1	AA3	1
AA3.2	AA3	2
AS1.1	AS1	1
AS1.2	AS1	2
AS1.3	AS1	3
AS2.1	AS2	1
AG1.1	AG1	1
AG1.2	AG1	2
AG1.3	AG1	3

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

RBS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
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, 2
HA5.1	HA5	1
HA6.1	HA6	1
HA7.1	HA7	1
HS1.1	HS1	1
HS6.1	HS6	1
HS7.1	HS7	1
HG7.1	HG7	1
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	1
SU4.2	SU3	2
SU5.1	SU4	1, 2, 3

RBS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
SU6.1	SU5	1
SU6.2	SU5	2
SU7.1	SU6	1, 2, 3
N/A	SU7	1, 2
SA1.1	SA1	1
SA3.1	SA2	1
SA6.1	SA5	1
SA8.1	SA9	1
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1
SG1.1	SG1	1
SG1.2	SG8	1

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

Category A – Abnormal Rad Levels / 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: A – 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:

AU1.1 Unusual Event

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

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

Table A-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Main Plant Vent - Primary	RE125	9.56E+08 μCi/sec	9.56E+07 μCi/sec	9.63E+06 μCi/sec	1.01E+05 μCi/sec
Gaseous	Main Plant Vent - Secondary	RE126	----	----	1.66E-01 μCi/ml	1.74E-03 μCi/ml
Gaseous	Fuel Bldg Vent - Primary	RE5A	7.75E+08 μCi/sec	7.75E+07 μCi/sec	7.75E+06 μCi/sec	6.50E+03 μCi/sec
Gaseous	Fuel Bldg Vent - Secondary	RE5B	----	----	1.72E-01μCi/ml	1.38E-03 μCi/ml
Gaseous	Radwaste Bldg Vent - Primary	RE6A	8.03E+08 μCi/sec	8.03E+07 μCi/sec	8.03E+06 μCi/sec	6.96E+04 μCi/sec
Gaseous	Radwaste Bldg Vent - Secondary	RE6B	----	----	2.12E-01 μCi/ml	1.71E-04 μCi/ml
Liquid	Liquid Radwaste	RE107	----	----	----	2 x Alarm Setpoint

Mode Applicability:

All

Definition(s):

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

Basis:

This IC addresses a potential ~~decrease~~ reduction 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~~as well as radioactivity releases that cause effluent radiation monitor readings to exceed 2 times the limit established by a radioactivity discharge permit. ~~This EAL will~~ Such releases are 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.

Reference(s):

1. SOP-0086 Digital Radiation Monitoring System
2. RSP-0008 Offsite Dose Calculation Manual
3. EP-CALC-RBS-1801 Radiological Effluent EAL Threshold Values
4. NEI 99-01 AU1

Category: A – 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:

AU1.2 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

None

Basis:

This IC addresses a potential ~~decrease~~ reduction 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 AA1A.

Reference(s):

1. RSP-0008 Offsite Dose Calculation Manual
2. NEI 99-01 AU1

Category: A – 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:

AA1.1 Alert

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

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

Note 4 The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table A-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Main Plant Vent - Primary	RE125	9.56E+08 μCi/sec	9.56E+07 μCi/sec	9.63E+06 μCi/sec	1.01E+05 μCi/sec
	Main Plant Vent - Secondary	RE126	----	----	1.66E-01 μCi/ml	1.74E-03 μCi/ml
	Fuel Bldg Vent - Primary	RE5A	7.75E+08 μCi/sec	7.75E+07 μCi/sec	7.75E+06 μCi/sec	6.50E+03 μCi/sec
	Fuel Bldg Vent - Secondary	RE5B	----	----	1.72E-01μCi/ml	1.38E-03 μCi/ml
	Radwaste Bldg Vent - Primary	RE6A	8.03E+08 μCi/sec	8.03E+07 μCi/sec	8.03E+06 μCi/sec	6.96E+04 μCi/sec
	Radwaste Bldg Vent - Secondary	RE6B	----	----	2.12E-01 μCi/ml	1.71E-04 μCi/ml
Liquid	Liquid Radwaste	RE107	----	----	----	2 x Alarm Setpoint

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem 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.

Reference(s):

1. SOP-0086 Digital Radiation Monitoring System
2. RSP-0008 Offsite Dose Calculation Manual
3. EP-CALC-RBS-1801 Radiological Effluent EAL Threshold Values
4. NEI 99-01 AA1

Category: A – 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:

AA1.2	**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 AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline.

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~~AS1.

Reference(s):

1. EIP-2-024 Offsite Dose Calculations
2. NEI 99-01 AA1

Category: A – 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:

AA1.3 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline.

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.~~

This EAL is assessed per the ODCM (ref. 2)

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

Reference(s):

1. EIP-2-024 Offsite Dose Calculations
2. RSP-0008 Offsite Dose Calculation Manual
3. NEI 99-01 AA1

Category: A – 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:

AA1.4 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline.

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~~AS1.

Reference(s):

1. EIP-2-014 Offsite Radiological Monitoring
2. NEI 99-01 AA1

Category: A – 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:

AS1.1 Site Area Emergency

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

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

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table A-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	Main Plant Vent - Primary	RE125	9.56E+08 µCi/sec	9.56E+07 µCi/sec	9.63E+06 µCi/sec	1.01E+05 µCi/sec
Gaseous	Main Plant Vent - Secondary	RE126	----	----	1.66E-01 µCi/ml	1.74E-03 µCi/ml
Gaseous	Fuel Bldg Vent - Primary	RE5A	7.75E+08 µCi/sec	7.75E+07 µCi/sec	7.75E+06 µCi/sec	6.50E+03 µCi/sec
Gaseous	Fuel Bldg Vent - Secondary	RE5B	----	----	1.72E-01µCi/ml	1.38E-03 µCi/ml
Gaseous	Radwaste Bldg Vent - Primary	RE6A	8.03E+08 µCi/sec	8.03E+07 µCi/sec	8.03E+06 µCi/sec	6.96E+04 µCi/sec
Gaseous	Radwaste Bldg Vent - Secondary	RE6B	----	----	2.12E-01 µCi/ml	1.71E-04 µCi/ml
Liquid	Liquid Radwaste	RE107	----	----	----	2 x Alarm Setpoint

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

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

Reference(s):

1. SOP-0086 Digital Radiation Monitoring System
2. RSP-0008 Offsite Dose Calculation Manual
3. EP-CALC-RBS-1801 Radiological Effluent EAL Threshold Values
4. NEI 99-01 AS1

Category: A – 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:

AS1.2	**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 AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline.

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.

Reference(s):

1. EIP-2-024 Offsite Dose Calculations
2. NEI 99-01 AS1

Category: A – 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:

AS1.3 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline.

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.

Reference(s):

1. EIP-2-014 Offsite Radiological Monitoring
2. NEI 99-01 AS1

Category: A – 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:

AG1.1 General Emergency

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

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

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table A-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	Main Plant Vent - Primary	RE125	9.56E+08 μCi/sec	9.56E+07 μCi/sec	9.63E+06 μCi/sec	1.01E+05 μCi/sec
Gaseous	Main Plant Vent - Secondary	RE126	----	----	1.66E-01 μCi/ml	1.74E-03 μCi/ml
Gaseous	Fuel Bldg Vent - Primary	RE5A	7.75E+08 μCi/sec	7.75E+07 μCi/sec	7.75E+06 μCi/sec	6.50E+03 μCi/sec
Gaseous	Fuel Bldg Vent - Secondary	RE5B	----	----	1.72E-01μCi/ml	1.38E-03 μCi/ml
Gaseous	Radwaste Bldg Vent - Primary	RE6A	8.03E+08 μCi/sec	8.03E+07 μCi/sec	8.03E+06 μCi/sec	6.96E+04 μCi/sec
Gaseous	Radwaste Bldg Vent - Secondary	RE6B	----	----	2.12E-01 μCi/ml	1.71E-04 μCi/ml
Liquid	Liquid Radwaste	RE107	----	----	----	2 x Alarm Setpoint

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

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

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

Reference(s):

1. SOP-0086 Digital Radiation Monitoring System
2. RSP-0008 Offsite Dose Calculation Manual
3. EP-CALC-RBS-1801 Radiological Effluent EAL Threshold Values
4. NEI 99-01 AG1

Category: A – 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:

AG1.2	**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 AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline.

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.~~

Reference(s):

1. EIP-2-024 Offsite Dose Calculations

2. NEI 99-01 AG1

Category: A – 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:

AG1.3 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline

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.~~

Reference(s):

1. EIP-2-014 Offsite Radiological Monitoring
2. NEI 99-01 AG1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: UNPLANNED loss of water level above irradiated fuel

EAL:

AU2.1 Unusual Event

UNPLANNED water level drop in the REFUELING PATHWAY as indicated by level instrumentation, low water level alarm or visual observation.

AND

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

- RMS-RE140 Refueling Floor Near North Entrance
- RMS-RE141 Refueling Floor Near South Entrance
- RMS-RE192 Fuel Building Operating Floor - South
- RMS-RE193 Fuel Building Operating Floor - North

Mode Applicability:

All

Definition(s):

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

REFUELING PATHWAY- Reactor cavity (well), containment spent fuel pool, fuel transfer canal, and fuel building spent fuel pools, but **not** including the reactor vessel, comprise the refueling pathway.

Basis:

This IC addresses a ~~decrease~~ drop 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~~ drop will be primarily determined by indications from available level instrumentation. Other sources of level indications may include reports from plant personnel (e.g., from a refueling crew) or video camera observations (if available). A significant drop in the water level may also cause a~~n~~ ~~increase~~ rise 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~~ rise 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.

The following low level alarms on P870 are related to low level in the REFUELING PATHWAY (ref. 4):

- SFP low alarm #0111 (H13-P870 / 56A / E02)
- Upper Transfer Pool Low alarm #0336 (H13-P870 / 56A / E03)
- Cask Pool Low alarm #0337 (H13-P870 / 56A / D03)
- Lower Transfer Pool low alarm #0335 (H13-P870 / 56A / F03)
- Rx Bldg Storage Pool low alarm #0112 (H13-P870 / 56A / H03)

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.

Reference(s):

1. AOP-0027 Fuel Handling Mishaps
2. USAR 12.3 Table 12.3-1 Area Direct Radiation Monitor Locations
3. USAR 9.1.4.2.3.11 Fuel Transfer System
4. ARP-870-0034 P870-56 Alarm Response
5. NEI 99-01 AU2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AA2.1 Alert

IMMINENT uncovery of irradiated fuel in the REFUELING PATHWAY

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY- The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the RBS ISFSI, Confinement Boundary is defined as the Holtec System Multi-Purpose Canister (MPC).

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

REFUELING PATHWAY- Reactor cavity (well), containment spent fuel pool, fuel transfer canal, and fuel building spent fuel pools, but **not** including the reactor vessel, comprise the refueling pathway.

Basis:

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

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

~~EAL #1~~

This EAL escalates from AU2.1 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 a~~n~~ ~~increase~~ rise in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.

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

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

~~EAL #3~~

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

Escalation of the emergency classification level would be via ICs AS1 ~~AS1or AS2 (*see AS2 Developer Notes*)~~.

Reference(s):

1. NEI 99-01 AA2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity

AND

High alarm on **any** Table A-2 radiation monitor

Table A-2 Fuel Damage Radiation Monitors
• RMS-RE140 Refueling Floor Near North Entrance • RMS-RE141 Refueling Floor Near South Entrance • RMS-RE192 Fuel Building Operating Floor - South • RMS-RE193 Fuel Building Operating Floor - North • RMS-RE5A(B) Fuel Building Ventilation Exhaust

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY- The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the RBS ISFSI, the Confinement Boundary is comprised the Holtec System Multi-Purpose Canister (MPC).

Basis:

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

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

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

~~ICs.~~

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

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

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

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

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

Reference(s):

1. AOP-0027 Fuel Handling Mishaps
2. USAR 12.3 Table 12.3-1 Area Direct Radiation Monitor Locations
3. USAR 12.3 Table 12.3-2 Airborne Process and Effluent Radiation Monitors
4. NEI 99-01 AA2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AA2.3	**Alert**
Lowering of spent fuel pool level to 108.0 ft. (Level 2) on SFC-LI29A/B	

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

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

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

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

~~This EAL addresses a release of radioactive material caused by mechanical damage to irradiated fuel. Damaging events may include the dropping, bumping or binding of an~~

~~assembly, or dropping a heavy load onto an assembly. A rise in readings on radiation monitors should be considered in conjunction with in-plant reports or observations of a potential fuel damaging event (e.g., a fuel handling accident).~~

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

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

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (SFC-LI29A/B) capable of identifying normal level (Level 1), SFP level approximately 23 ft. above the top of the fuel racks, (Level 2) 107 ft. 10 5/16 in. (rounded to 108.0 ft.) which is that level adequate to provide substantial radiation shielding for a person standing on the SFP operating deck, and SFP level at the top of the fuel racks (Level 3) 85 ft. 10 5/16 in. (rounded to 86.0 ft.) (ref. 1). RBS uses a Level 3 of approximately one foot above the highest point of any fuel rack providing added margin.

Spent Fuel Pool Level indicators SFC-LI29A and B are read on the 98 ft. elevation Control Building on the interior of the West exterior wall (ref. 2).

Reference(s):

1. RBG-47570 Completion of Required Action by NRC Order EA-12-051 Reliable SFP Instrumentation
2. RBS-FSG-011 Alternate Spent Fuel Pool Makeup and Cooling
3. NEI 99-01 AA2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AS2.1	**Site Area Emergency**
Lowering of spent fuel pool level to 86.0 ft. (Level 3) on SFC-LI29A/B	

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

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

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (SFC-LI29A/B) capable of identifying normal level (Level 1), SFP level approximately 23 ft. above the top of the fuel racks, (Level 2) 107 ft. 10 5/16 in. (rounded to 108.0 ft.) which is that level adequate to provide substantial radiation shielding for a person standing on the SFP operating deck, and SFP level at the top of the fuel racks (Level 3) 85 ft. 10 5/16 in. (rounded to 86.0 ft.) (ref. 1). RBS uses a Level 3 of approximately one foot above the highest point of any fuel rack providing added margin.

Spent Fuel Pool Level indicators SFC-LI29A and B are read on the 98 ft. elevation Control Building on the interior of the West exterior wall (ref. 2).

Reference(s):

1. RBG-47570 Completion of Required Action by NRC Order EA-12-051 Reliable SFP Instrumentation
2. RBS-FSG-011 Alternate Spent Fuel Pool Makeup and Cooling
3. NEI 99-01 AS2

Category: A – 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:

AG2.1 General Emergency

Spent fuel pool level **cannot** be restored to at least 86.0 ft. (Level 3) on SFC-LI29A/B for ≥ 60 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (SFC-LI29A/B) capable of identifying normal level (Level 1), SFP level approximately 23 ft. above the top of the fuel racks, (Level 2) 107 ft. 10 5/16 in. (rounded to 108.0 ft.) which is that level adequate to provide substantial radiation shielding for a person standing on the SFP operating deck, and SFP level at the top of the fuel racks (Level 3) 85 ft. 10 5/16 in. (rounded to 86.0 ft.) (ref. 1). RBS uses a Level 3 of approximately one foot above the highest point of any fuel rack providing added margin.

Spent Fuel Pool Level indicators SFC-LI29A and B are read on the 98 ft. elevation Control Building on the interior of the West exterior wall (ref. 2).

Reference(s):

1. RBG-47570 Completion of Required Action by NRC Order EA-12-051 Reliable SFP Instrumentation
2. RBS-FSG-011 Alternate Spent Fuel Pool Makeup and Cooling
3. NEI 99-01 AG2

Category: A – 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:

AA3.1 Alert

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

- Control Room (RMS-RE170)
- Central Alarm Station (by survey)

Mode Applicability:

All

Definition(s):

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

Basis:

Areas that meet this threshold include the Control Room (CR) and the Central Alarm Station (CAS). The Control Room is monitored for excessive radiation by one detector, RMS-RE170 (ref. 1). The CAS is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations. There are no permanently installed area radiation monitors in CAS that may be used to assess this EAL threshold. Therefore, this threshold is evaluated using local radiation survey for this area.

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~~ rise in 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, C or F ICs.

Reference(s):

1. USAR 12.3 Table 12.3-1 Area Direct Radiation Monitor Locations
2. NEI 99-01 AA3

Category: A – 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:

AA3.2 Alert

An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table A-3 room or area (Note 5)

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

Table A-3 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode**
Auxiliary Building 70' RHR B Pump Room	3
Auxiliary Building 80' RHR A Pump Room	3
Auxiliary Building 114' West	3
Control Building 95' Div 1 RSS Room	3

Mode Applicability:

3 – Hot Shutdown

Definition(s):

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

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

Basis:

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

For ~~EAL #2~~ AA3.2, 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~~ higher 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~~ rise occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode ~~4~~3.
- The ~~increased~~ higher radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).
- ~~The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).~~
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or IMPEDE a required action.

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

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).

EAL AA3.2 mode applicability has been limited to the mode limitations of Table A-3 (Mode 3 only).

Reference(s):

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

Category C – Cold Shutdown / Refueling System Malfunction

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

Category C EALs are directly associated with cold shutdown or refueling system safety functions. Given the variability of plant configurations (e.g., systems out-of-service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The cold shutdown and refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, CONTAINMENT CLOSURE, and fuel clad integrity for the applicable operating modes (4 - Cold Shutdown, 5 - Refueling, DEF – Defueled).

The events of this category pertain to the following subcategories:

1. RPV Level

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

2. Loss of Emergency AC Power

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

3. RCS Temperature

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

4. Loss of Vital DC Power

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

5. Loss of Communications

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

6. Hazardous Event Affecting Safety Systems

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

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: UNPLANNED loss of RPV inventory

EAL:

CU1.1 Unusual Event

UNPLANNED loss of reactor coolant results in RPV water 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

Basis:

River Bend is equipped with multiple RPV water level instruments including: Wide Range, Fuel Zone, Shutdown Range, Upset Range, and Narrow Range (ref. 1). Multiple instruments on different reference and variable legs should be monitored. The Upset Range and Shutdown Range instruments share a common reference leg; therefore, Narrow Range instruments should be routinely monitored when relying on Shutdown or Upset Range instrument as the primary indication.

With the plant in Cold Shutdown, RPV water level is normally maintained above the RPV low level scram setpoint of 9.7 in. (ref. 2). However, if RPV level is being controlled below the RPV low level scram 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, RPV water level is normally maintained at or above the reactor vessel flange. Technical Specifications require at least 23 ft of water above the top of the reactor vessel flange in the refueling cavity during refueling operations (ref. 3). The RPV flange is at approximately 200 in. on the Shutdown Range. (ref. 4).

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

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

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

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

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

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

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

Reference(s):

1. EOP-1 RPV Control
2. Technical Requirements Table 3.3.1.1-1 RPS Instrumentation
3. Technical Specification 3.9.6 Reactor Pressure Vessel (RPV) Water Level - Irradiated Fuel
4. GMP-0102 Reactor Vessel Disassembly
5. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: UNPLANNED loss of RPV inventory

EAL:

CU1.2 Unusual Event

RPV water level **cannot** be monitored

AND EITHER

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

Table C-1 Sumps/Pool
• Drywell equipment drain sump • Drywell floor drain sump • Pedestal floor drain sump • CTMT equipment drain sump • CTMT floor drain sump • Suppression Pool • RHR A, B, C, HPCS, LPCS, RCIC room sumps • Auxiliary Building floor drain sump

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

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

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

Basis:

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

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Rise in drywell equipment drain sump level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 1, 2). An Auxiliary Building sump level rise may also be indicative of RCS inventory losses external to the Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

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

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

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

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

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

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

Reference(s):

1. SOP-0104 Floor and Equipment Drains System
2. SOP-0033 Drywell and Containment Leak Detection System
3. EOP-3 Secondary Containment and Radioactivity Release Control
4. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Significant Loss of RPV inventory

EAL:

CA1.1	**Alert**
Loss of RPV inventory as indicated by RPV water level < -43 in. (Level 2)	

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

None

Basis:

The threshold RPV water level of -43 in. is the Level 2 actuation setpoint for HPCS and RCIC. Although RCIC cannot restore RPV inventory in the cold condition, the Level 2 actuation setpoint is operationally significant and is indicative of a loss of RPV inventory significantly below the low RPV water level scram setpoint specified in CU1.1 (ref. 1, 2).

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

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

Although related, this EAL~~#1~~ is concerned with the loss of RPV 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). A~~n~~ ~~increase~~ rise in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

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

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

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

Reference(s):

1. Technical Requirements Table 3.3.5.1-1, Emergency Core Cooling System Instrumentation
2. Technical Requirements Table 3.3.5.2-1, RCIC Instrumentation
3. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Significant Loss of RPV inventory

EAL:

CA1.2 Alert

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

AND EITHER

- UNPLANNED rise in **any** Table C-1 sump or pool levels due to a loss of RPV inventory
- Visual observation of UNISOLABLE RCS leakage

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table C-1 Sumps/Pool
• Drywell equipment drain sump • Drywell floor drain sump • Pedestal floor drain sump • CTMT equipment drain sump • CTMT floor drain sump • Suppression Pool • RHR A, B, C, HPCS, LPCS, RCIC room sumps • Auxiliary Building floor drain sump

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

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

Basis:

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

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Rise in drywell equipment drain sump level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 1, 2). An Auxiliary Building sump level rise may also be indicative of RCS inventory losses external to the Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

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

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

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

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

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

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

Reference(s):

1. SOP-0104 Floor and Equipment Drains System
2. SOP-0033 Drywell and Containment Leak Detection System
3. EOP-3 Secondary Containment and Radioactivity Release Control
4. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.1 Site Area Emergency

CONTAINMENT CLOSURE **not** established

AND

RPV water level < -143 in. (Level 1)

Mode Applicability:

4 – Cold Shutdown, 5 – Refueling

Definition(s):

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

Containment Closure is established when the containment requirements of OSP-0037 are met and at least one integral barrier to the release of radioactive material is provided, within the specified limits using STP-057-3804.

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

Basis:

The threshold RPV water level of -143 in. is the low-low-low ECCS actuation setpoint (Level 1). The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further lowering of RPV water level and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier (ref. 1, 2).

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

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If ~~RCS/reactor vessel~~RPV 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~~RPV levels of EALs CS1.1 and CS1.2 ~~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, 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]).~~

Thi~~ese~~s EAL~~s~~ address~~es~~ 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.

Reference(s):

1. Technical Requirements Table 3.3.5.1-1, Emergency Core Cooling System Instrumentation
2. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.2	**Site Area Emergency**

CONTAINMENT CLOSURE established

AND

RPV water level < -162 in. (TAF)

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

Containment Closure is established when the containment requirements of OSP-0037 are met and at least one integral barrier to the release of radioactive material is provided, within the specified limits using STP-057-3804.

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

Basis:

When RPV level drops to the top of active fuel (TAF) (an indicated RPV level of -162 in.), core uncovery starts to occur (ref. 1).

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

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV 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~~RPV levels of EALs ~~1.b and 2.b~~ CS1.1 and CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

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

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

Reference(s):

1. EPSTG*0002 Appendix B EOP and SAP Bases
2. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.3 Site Area Emergency

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

AND

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

- UNPLANNED rise in **any** Table C-1 sump or pool levels of sufficient magnitude to indicate core uncovery
- Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery
- RMS-RE140 Refueling Floor Near North Entrance, RMS-RE141 Refueling Floor Near South Entrance or RMS-RE16 A/B Primary containment - PAM A/B reading > 9 R/hr

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table C-1 Sumps/Pool
• Drywell equipment drain sump • Drywell floor drain sump • Pedestal floor drain sump • CTMT equipment drain sump • CTMT floor drain sump • Suppression Pool • RHR A, B, C, HPCS, LPCS, RCIC room sumps • Auxiliary Building floor drain sump

Mode Applicability:

4 – Cold Shutdown, 5 – Refueling

Definition(s):

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

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

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

Basis:

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

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Rise in drywell equipment drain sump level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 1, 2). An Auxiliary Building sump level rise may also be indicative of RCS inventory losses external to the Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

In the Refueling Mode, as water level in the RPV lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in indications on area radiation monitors. The 9 R/hr value is selected for this EAL because it is 90% of the scale for RMS-RE140 and 141 (lower range monitors) and on scale for the higher range monitors. This value represents a reading that is higher than that likely to be attributable to normal refuel floor operations. These monitors are located in the Containment on the refuel floor.

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

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

~~Outage/shutdown contingency plans typically provide for re-establishing or verifying~~

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

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

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

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

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

Reference(s):

1. SOP-0104 Floor and Equipment Drains System
2. SOP-0033 Drywell and Containment Leak Detection System
3. EOP-3 Secondary Containment and Radioactivity Release Control
4. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CG1.1 General Emergency

RPV level < -162 in. (TAF) for ≥ 30 min. (Note 1)

AND

Any Containment Challenge indication, Table C-2

Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Table C-2 Containment Challenge Indications

- CONTAINMENT CLOSURE **not** established (Note 6)
- Drywell or containment hydrogen concentration > 4%
- UNPLANNED rise in containment pressure
- Exceeding one or more Secondary Containment Control MAX SAFE area radiation levels:

Area	DRMS Grid 2	Max. Safe Operating Value
RHR Equip Rm A	1213	9.5E+03 mR/hr
RHR Equip Rm B	1214	9.5E+03 mR/hr
RHR Equip Rm C	1215	9.5E+03 mR/hr

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

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

Containment Closure is established when the containment requirements of OSP-0037 are met and at least one integral barrier to the release of radioactive material is provided, within the specified limits using STP-057-3804.

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

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

Basis:

When RPV level drops below -162 in., core uncovery starts to occur (ref. 1).

Four conditions are associated with a challenge to Containment integrity:

- CONTAINMENT CLOSURE is not established.
- 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 hydrogen concentration has exceeded the minimum necessary to support a hydrogen burn (4%). The Igniter System is designed to prevent hydrogen accumulation by locally burning hydrogen in a controlled manner as soon as the hydrogen enters the containment atmosphere and reaches the igniters. For high rates of hydrogen production, ignition occurs at the lowest concentration that can support ignition. Following ignition, hydrogen is consumed through formation of diffusion flames where the gas enters the containment, thus controlling hydrogen concentration at approximately 4% (ref. 2).
- Any UNPLANNED rise in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of CONTAINMENT CLOSURE capability. UNPLANNED containment pressure rise indicates CONTAINMENT CLOSURE cannot be assured and the containment cannot be relied upon as a barrier to fission product release.
- Secondary Containment radiation monitors should provide indication of a larger release that may be indicative of a challenge to CONTAINMENT CLOSURE. The MAX SAFE radiation levels are indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in Table SC-2 of EOP-3, Secondary Containment

and Radioactivity Release Control that are in service under Cold Shutdown conditions (ref. 3).

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

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV 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.~~

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

Reference(s):

1. EOP-1 RPV Control
2. EPSTG*0002 Appendix B EOP and SAP Bases

3. EOP-3 Secondary Containment and Radioactivity Release Control
4. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CG1.2 General Emergency

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

AND

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

- UNPLANNED rise in **any** Table C-1 sump or pool levels of sufficient magnitude to indicate core uncovery
- Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery
- RMS-RE140 Refueling Floor Near North Entrance, RMS-RE141 Refueling Floor Near South Entrance or RMS-RE16 A/B Primary containment - PAM A/B reading > 9 R/hr

AND

Any Containment Challenge indication, Table C-2

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Table C-1 Sumps/Pool
• Drywell equipment drain sump • Drywell floor drain sump • Pedestal floor drain sump • CTMT equipment drain sump • CTMT floor drain sump • Suppression Pool • RHR A, B, C, HPCS, LPCS, RCIC room sumps • Auxiliary Building floor drain sump

Table C-2 Containment Challenge Indications

- CONTAINMENT CLOSURE **not** established (Note 6)
- Drywell or containment hydrogen concentration > 4%
- UNPLANNED rise in containment pressure
- Exceeding one or more Secondary Containment Control MAX SAFE area radiation levels:

Area	DRMS Grid 2	Max. Safe Operating Value
RHR Equip Rm A	1213	9.5E+03 mR/hr
RHR Equip Rm B	1214	9.5E+03 mR/hr
RHR Equip Rm C	1215	9.5E+03 mR/hr

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

Containment Closure is established when the Containment requirements of OSP-0037 (ref. 4.1.12) are met with the following exception: a functional barrier must exist at the time of the event (i.e., cannot rely on contingency methods to establish a functional barrier).

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

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

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

Basis:

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

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Rise in drywell equipment drain sump level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 1, 2). An Auxiliary Building sump level rise may also be indicative of RCS inventory losses external to the Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

In the Refueling Mode, as water level in the RPV lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in indications on area radiation monitors. The 9 R/hr value is selected for this EAL because it is 90% of the scale for RMS-RE140 and 141 (lower range monitors) and on scale for the higher range monitors. This value represents a reading that is higher than that likely to be attributable to normal refuel floor operations. These monitors are located in the Containment on the refuel floor.

Four conditions are associated with a challenge to Containment integrity:

- CONTAINMENT CLOSURE is not established.
- 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 hydrogen concentration has exceeded the minimum necessary to support a hydrogen

burn (4%). The Igniter System is designed to prevent hydrogen accumulation by locally burning hydrogen in a controlled manner as soon as the hydrogen enters the containment atmosphere and reaches the igniters. For high rates of hydrogen production, ignition occurs at the lowest concentration that can support ignition. Following ignition, hydrogen is consumed through formation of diffusion flames where the gas enters the containment, thus controlling hydrogen concentration at approximately 4% (ref. 3).

- Any UNPLANNED rise in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of CONTAINMENT CLOSURE capability. UNPLANNED containment pressure rise indicates CONTAINMENT CLOSURE cannot be assured and the containment cannot be relied upon as a barrier to fission product release.
- Secondary Containment radiation monitors should provide indication of a larger release that may be indicative of a challenge to CONTAINMENT CLOSURE. The MAX SAFE radiation levels are indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in Table SC-2 of EOP-3, Secondary Containment and Radioactivity Release Control that are in service under Cold Shutdown conditions (ref. 4).

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

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

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

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

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

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

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

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

Reference(s):

1. SOP-0104 Floor and Equipment Drains System
2. SOP-0033 Drywell and Containment Leak Detection System
3. EPSTG*0002 Appendix B EOP and SAP Bases
4. EOP-3 Secondary Containment and Radioactivity Release Control
5. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CU2.1 Unusual Event

AC power capability, Table C-3, to DIV I and DIV II 4.16 KV ENS buses 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table C-3 AC Power Sources
Offsite • 1RTX-XSR1C • 1RTX-XSR1D
Onsite • EGS-EG1A • EGS-EG1B

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, DEF - Defueled

Definition(s):

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

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

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

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

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

Basis:

The DIV III bus (1E22*S004) is not credited because it only supplies power to the HPCS pump and associated loads, not any long term decay heat removal systems. In particular, suppression pool cooling mechanisms would be essential subsequent to a station blackout.

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~~ greater 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 ENS bus. Some examples of this condition are presented below.

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

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

Reference(s):

1. USAR Section 8.1 Electric Power Introduction
2. EE-001AC Startup Electrical Distribution Chart
3. SOP-0046 4.16 KV System
4. AOP-0004 Loss of Offsite Power
5. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CA2.1 Alert

Loss of **all** offsite and **all** onsite AC power capability to DIV I and DIV II 4.16 KV ENS buses for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, DEF - Defueled

Definition(s):

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

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

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

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

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

Basis:

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.

Mitigative strategies using other power sources (Division III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as a Site Area Emergency because of the ~~increased~~ greater time available to restore an ~~ESF~~ ENS 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.

This EAL is the cold condition equivalent of the hot condition EAL SS1.1.

Reference(s):

1. USAR Section 8.1 Electric Power Introduction
2. EE-001AC Startup Electrical Distribution Chart
3. SOP-0046 4.16 KV System
4. AOP-0004 Loss of Offsite Power
5. NEI 99-01 CA2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED rise in RCS temperature

EAL:

CU3.1	**Unusual Event**
UNPLANNED rise in RCS temperature to > 200°F due to loss of decay heat removal capability	

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

Containment Closure is established when the containment requirements of OSP-0037 are met and at least one integral barrier to the release of radioactive material is provided, within the specified limits using STP-057-3804.

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

Basis:

This IC addresses an UNPLANNED ~~increase~~ rise in RCS temperature above the Technical Specification cold shutdown temperature limit, ~~or the inability to determine RCS temperature and level,~~ and represents a potential degradation of the level of safety of the plant. If the ~~RCS~~ RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Director should also refer to ~~IC~~ EAL CA3.1.

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

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

During an outage, the level in the reactor vessel will normally be maintained ~~at or~~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~~ rise 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.

Reference(s):

1. Technical Specification Table 1.1-1 Modes
2. STP-050-0700 RCS Pressure and Temperature Limits Verification
3. AOP-0051 Loss of Decay Heat Removal
4. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED rise in RCS temperature

EAL:

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5- Refueling

Definition(s):

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

Containment Closure is established when the containment requirements of OSP-0037 are met and at least one integral barrier to the release of radioactive material is provided, within the specified limits using STP-057-3804.

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

Basis:

This ~~IC~~ EALEAL addresses ~~an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit, or~~ the inability to determine RCS temperature and level, andand 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.

Reference(s):

1. EOP-1 RPV Control
2. Technical Specifications Table 1.1-1 Modes
3. STP-050-0700 RCS Pressure and Temperature Limits Verification
4. AOP-0051 Loss of Decay Heat Removal
5. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.1 Alert

UNPLANNED rise in RCS temperature to > 200°F for > Table C-4 duration (Note 1)

OR

UNPLANNED RPV pressure rise > 10 psig

Note 1: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

Table C-4 RCS Heat-up Duration Thresholds

RCS Status	CONTAINMENT CLOSURE Status	Heat-up Duration
Intact	N/A	60 min.*
Not intact	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:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

Containment Closure is established when the containment requirements of OSP-0037 are met and at least one integral barrier to the release of radioactive material is provided, within the specified limits using STP-057-3804.

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

Basis:

In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the RCS pressure rise criteria when the RCS is intact in Mode 4 or based on time to boil data when in Mode 5 or the RCS is not intact in Mode 4.

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

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

The RCS Heat-up Duration Thresholds table addresses a~~n~~ ~~increase~~ rise 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~~rise.

The RCS Heat-up Duration Thresholds table also addresses a~~n~~ ~~increase~~ rise 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~~ rise without a substantial degradation in plant safety.

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

The RCS pressure rise threshold ~~EAL #2~~ 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.

Reference(s):

1 Technical Specifications Table 1.1-1 Mods
2. STP-050-0700 RCS Pressure and Temperature Limits Verification
3. AOP-0051 Loss of Decay Heat Removal
4. NEI 99-01 CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – Loss of Vital DC Power

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

EAL:

CU4.1 Unusual Event

Indicated voltage is < 105 VDC on required Safety Related DIV I and DIV II 125 VDC buses for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

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

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

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

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

Basis

Safety Related DC buses ENB-SWG01A (DIV I) and ENB-SWG01B (DIV II) feed the Division I and Division II loads respectively. The Division I and Division II batteries each have 60 cells with a specific minimum voltage of 1.75 volts/cell. These cell voltages yield minimum design bus voltages of 105 VDC (ref. 1).

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

As used in this EAL, "required" means the vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if

Train A is out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an Unusual Event. A loss of Vital DC power to Train A would not warrant an emergency classification.

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

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

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

Reference(s):

1. Safety Related Battery Specification 244.521
2. USAR 8.3.2 DC Power Systems
3. NEI 99-01 CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.1 Unusual Event

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

OR

Loss of **all** Table C-5 State and local agency communication methods

OR

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

Table C-5 Communication Methods

System	Onsite	State/ Local	NRC
Plant radio system	X		
Plant Paging System	X		
Sound powered phones	X		
In-plant telephones	X		
Emergency Notification System (ENS)			X
Commercial Telephone System		X	X
Satellite Phones		X	X
State of Louisiana Radio		X	
State and Local Hotline radio		X	
INFORM Notification System		X	

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, DEF – Defueled

Definition(s):

None

Basis:

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

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

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

The second EAL condition ~~EAL #2~~ addresses a total loss of the communications methods used to notify all ~~OROs~~ State and local agencies of an emergency declaration. The ~~OROs~~ State and local agencies referred to here are ~~(see Developer Notes)~~the Louisiana Department of Environmental Quality, Governor's Office of Homeland Security and Emergency Preparedness, five Local Parishes Office of Homeland Security and Emergency Preparedness and 24 hour notification points, Mississippi Emergency Management Agency and the Mississippi Highway Patrol.

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

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

Reference(s):

1. RBS Emergency Plan Section 13.3.6.1.5.4 Communications
2. RBS Emergency Plan Section 13.3.6.2.1 Site Communications
3. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting Safety Systems

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

EAL:

CA6.1 Alert

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

AND

Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode

AND EITHER:

- Event damage has caused indications of degraded performance to the second train of the SAFETY SYSTEM needed for the current operating mode
- Event damage has resulted in VISIBLE DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode

(Notes 9, 10)

Note 9: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then emergency classification is **not** warranted.

Note 10: If the hazardous event **only** resulted in VISIBLE DAMAGE, with **no** indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is **not** warranted.

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

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

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

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

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

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

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

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

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

VISIBLE DAMAGE - Damage to a SAFETY SYSTEM train that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected SAFETY SYSTEM train.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance;

commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

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

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

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

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

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

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

This EAL is the cold condition equivalent of hot condition EAL SA8.1.

Reference(s):

1. EP FAQ 2016-002
2. NEI 99-01 CA6

Category E – Independent Spent Fuel Storage Installation (ISFSI)

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

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

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

The RBS ISFSI is located wholly within the plant PROTECTED AREA. Therefore any security event related to the ISFSI are classified under Category H1 security event related EALs.

Category: ISFSI

Subcategory: Confinement Boundary

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Unusual Event

Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask (HI-STORM overpack) > **EITHER** of the following:

- 60 mRem/hr ($\gamma + \eta$) on the top of the overpack
- 600 mRem/hr ($\gamma + \eta$) on side of the overpack (excluding inlet and outlet ducts)

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY- The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the RBS ISFSI, Confinement Boundary is defined as the Holtec System Multi-Purpose Canister (MPC).

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI): A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

Basis:

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

The existence of "damage" is determined by radiological survey. The specified EAL threshold values correspond to 2 times the cask technical specification values. The technical specification (licensing bases document) multiple of "2 times", which is also used in Recognition Category A IC AU1, is used here to distinguish between non-emergency and emergency conditions (ref. 2). The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

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

Reference(s):

1. USAR 9.1.2.5 Holtec HI-STORM Dry Spent Fuel Storage System
2. RBS HI-STORM 100 SYSTEM Certificate of Compliance for Spent Fuel Storage Casks Amendment 5, Appendix A Technical Specifications for the HI-STORM 100 Cask System Section 5.7.4
3. NEI 99-01 E-HU1

Category F – Fission Product Barrier Degradation

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

EALs in this category represent threats to the defense in depth design concept that precludes the release of highly radioactive fission products to the environment. This concept relies on multiple physical barriers any one of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment. The primary fission product barriers are:

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

B. Reactor Coolant System Barrier (RCB): The RCS Barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping out to and including the isolation valves.

C. Containment Barrier (CNB): The Containment Barrier includes the drywell, the containment, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. Containment Barrier thresholds are used as criteria for escalation of the Emergency Classification Level (ECL) from an Alert to a Site Area Emergency or a General Emergency.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1. "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

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

Site Area Emergency:
Loss or potential loss of any two barriers

General Emergency:
Loss of any two barriers and loss or potential loss of third barrier

The logic used for emergency classification based on fission product barrier monitoring should reflect the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier.

- Unusual Event ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.
- For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC AG1 has been exceeded.
- The fission product barrier thresholds specified within a scheme reflect plant-specific RBS design and operating characteristics.
- As used in this category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside the primary containment, an interfacing system, or outside of the primary containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered to be RCS leakage.
- At the Site Area Emergency level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the 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: **Any** loss or **any** potential loss of either Fuel Clad or RCS

EAL:

FA1.1	**Alert**
Any loss or **any** potential loss of either Fuel Clad or RCS barrier (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds, bases and references.

At the Alert classification level, Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1

Reference(s):

1. NEI 99-01 FA1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss or potential loss of **any** two barriers

EAL:

FS1.1	**Site Area Emergency**
Loss or potential loss of **any** two barriers (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds, bases and references.

At the Site Area Emergency classification level, each barrier is weighted equally. A Site Area Emergency is therefore appropriate for any combination of the following conditions:

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

At the Site Area Emergency classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, the Emergency Director would have greater assurance that escalation to a General Emergency is less IMMINENT.

Reference(s):

1. NEI 99-01 FS1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss of **any** two barriers and loss or potential loss of third barrier

EAL:

FG1.1 General Emergency

Loss of **any** two barriers

AND

Loss or potential loss of the third barrier (Table F-1)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds, bases and references.

At the General Emergency classification level each barrier is weighted equally. A General Emergency is therefore appropriate for any combination of the following conditions:

- Loss of Fuel Clad, RCS and Containment Barriers
- Loss of Fuel Clad and RCS Barriers with potential loss of Containment Barrier
- Loss of RCS and Containment Barriers with potential loss of Fuel Clad Barrier
- Loss of Fuel Clad and Containment Barriers with potential loss of RCS Barrier

Reference(s):

1. NEI 99-01 FG1

Table F-1 Fission Product Barrier Threshold Matrix & Bases

Table F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Containment). The table is structured so that each of the three barriers occupies adjacent columns. Each fission product barrier column is further divided into two columns; one for Loss thresholds and one for Potential Loss thresholds.

The first column of the table (to the left of the Fuel Clad Barrier Loss column) lists the categories (types) of fission product barrier thresholds. The fission product barrier categories are:

A. RPV Water Level

B. RCS Leak Rate

C. Containment Conditions

D. Containment Radiation / RCS Activity

E. Containment Integrity or Bypass

F. 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 barrier column beginning with number one (ex., FCB1, FCB2...FCB6).

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 to category Loss followed by category Potential Loss beginning with Category A, then B,…,F.

Table F-1 Fission Product Barrier Threshold Matrix						
	Fuel Clad Barrier (FCB)		**Reactor Coolant System Barrier (RCB)**		**Containment Barrier (CNB)**	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A **RPV Water Level**	FCB1 SAP entry is required	FCB2 RPV water level **cannot** be restored and maintained > -162 in. (TAF) or **cannot** be determined	RCB1 RPV water level **cannot** be restored and maintained > -162 in. (TAF) or **cannot** be determined	None	None	CNB1 SAP entry is required
B **RCS Leak Rate**	None	None	RCB2 UNISOLABLE break in **any** of the following: • Main steam lines • RCIC steam Line • RWCU • Feedwater RCB3 Emergency Depressurization is required	RCB4 UNISOLABLE primary system leakage that results in exceeding **EITHER**: • One or more EOP-3 Max Normal area radiation operating value (Table F-2) • One or more Isolation Temperature alarms (Table F-2)	CNB2 UNISOLABLE primary system leakage that results in exceeding **EITHER**: • One or more EOP-3 Max Safe area radiation operating value that can be read in the Control Room (Table F-2) • One or more EOP-3 Max Safe area temperature operating value (Table F-2)	None
C **CTMT Conditions**	None	None	RCB5 Drywell pressure > 1.68 psid due to RCS leakage	None	CNB3 UNPLANNED rapid drop in containment pressure following containment pressure rise CNB4 Containment pressure response **not** consistent with LOCA conditions	CNB5 Containment pressure > 15 psig CNB6 Drywell or containment hydrogen concentration > 4% CNB7 Parameters **cannot** be restored and maintained within the safe zone of the HCTL curve (EOP Figure 2)
D **CTMT Rad / RCS Activity**	FCB3 Containment radiation (RMS-RE16) > 3,000 R/hr FCB4 Coolant activity > 300 µCi/gm dose equivalent I-131	None	RCB6 Drywell radiation (RMS-RE20) > 30 R/hr	None	None	CNB8 Containment radiation (RMS-RE16) > 12,000 R/hr
E **CTMT Integrity or Bypass**	None	None	None	None	CNB9 UNISOLABLE direct downstream pathway to the environment exists after Containment isolation signal CNB10 Intentional Containment venting per EOPs	None
F **Emergency Director Judgment**	FCB5 **Any** condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier	FCB6 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier	RCB7 **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	RCB8 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	CNB11 **Any** condition in the opinion of the Emergency Director that indicates loss of the Containment barrier	CNB12 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier

Barrier: Fuel Clad

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

FCB1
SAP entry is required

Definition(s):

None

Basis:

Emergency Operating Procedure (EOPs) specify entry to the Severe Accident Procedures (SAPs) when core cooling is severely challenged. These EOPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined (ref. 1, 2).

The EOP conditions requiring SAP entry represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

This threshold is also a Potential Loss of the Containment barrier (CNB1). Since SAP entry occurs after core uncovery has occurred a Loss of the RCS barrier exists (RCB1). SAP entry, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

The Loss threshold represents the EOP requirement for entry into the SAPs. This is identified in the BWROG EPGs/SAGs when adequate core cooling cannot be assured.~~The Loss threshold represents the EOP requirement for primary containment flooding. This is identified in the BWROG EPGs/SAGs when the phrase, "Primary Containment Flooding Is Required," appears. Since a site-specific RPV water level is not specified here, the Loss threshold phrase, "Primary containment flooding required," also accommodates the EOP need to flood the primary containment when RPV water level cannot be determined and core damage due to inadequate core cooling is believed to be occurring.~~

Reference(s):

1. EOP-1 RPV Control
2. EOP-4 RPV Flooding
3. EP FAQ 2015-004
4. NEI 99-01, RPV Water Level Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

FCB2
RPV water level **cannot** be restored and maintained > -162 in. (TAF) or **cannot** be determined

Definition(s):

None

Basis:

An RPV water level instrument reading of -162 in. indicates RPV level is at the top of active fuel (TAF) (ref. 1). When RPV level is at or above the TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV water level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops to TAF, the level is indicative of a challenge to core cooling and the Fuel Clad barrier.

When RPV water level cannot be determined, EOPs require entry to EOP-4, RPV Flooding. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained (ref. 2). When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-4 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in scram-failure events). If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the fuel clad barrier exists.

Note that EOP-1A, RPV Control, ATWS, may require intentionally lowering RPV water level to -162 in. and control level between the Minimum Steam Cooling RPV Water Level (MSCRWL) and the top of active fuel (ref. 3). Under these conditions, a high-power ATWS event exists and requires at least a Site Area Emergency classification in accordance with the System Malfunction - RPS Failure EALs.

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ~~ICs~~ EALs ~~SA5~~ SA6.1 or ~~SS5~~ SS6.1 will dictate the need for emergency classification.

This water level corresponds to the top of the active fuel and is used in the EOPs to indicate a challenge to core cooling.

The RPV water level threshold is the same as RCS barrier Loss threshold ~~2.A~~RCB1. Thus, this threshold indicates a Potential Loss of the Fuel Clad barrier and a Loss of the RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

This threshold is considered to be exceeded when, as specified in the site-specific EOPs, RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this Fuel Clad barrier Potential Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

The term "cannot be restored and maintained above" means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value cannot be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active fuel, but does not permit extended operation below the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

Since the loss of ability to determine if adequate core cooling is being provided presents a significant challenge to the fuel clad barrier, a potential loss of the fuel clad barrier is specified.

Reference(s):

1. EOP-1 RPV Control
2. EOP-4 RPV Flooding
3. EOP-1A RPV Control, ATWS
4. NEI 99-01 RPV Water Level Potential Loss 2.A

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

FCB3

Containment radiation (RMS-RE16) > 3,000 R/hr

Definition(s):

None

Basis:

The containment 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%~~significant fuel clad damage (ref. 1). Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold ~~C.1~~RCB6 since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the ECL to a Site Area Emergency.

There is no Fuel Clad barrier Potential Loss threshold associated with ~~RCS Activity / Containment~~ CTMT Radiation/RCS Activity.

Reference(s):

1. Calculation G13.18.9.4-045 Containment Doses for Emergency Action Levels (EALs)
2. NEI 99-01 Primary Containment Radiation Fuel Clad Loss 4.A

Barrier: Fuel Clad

Category: D. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

FCB4
Coolant activity > 300 µCi/gm dose equivalent I-131

Definition(s):

None

Basis:

This threshold indicates that RCS radioactivity concentration is greater than 300 µCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications.

There is no Fuel Clad barrier Potential Loss threshold associated with ~~RCS Activity / Containment~~CTMT Radiation/RCS Activity.

Reference(s):

1. NEI 99-01 RCS Activity Fuel Clad Loss 1.A

Barrier: Fuel Clad

Category: D. CTMT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

FCB5
Any condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost.

Reference(s):

1. NEI 99-01 Emergency Director Judgment Fuel Clad Loss 6.A

Barrier: Fuel Clad

Category: F. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

FCB6
Any condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that are to be used by the Emergency Director 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.

Reference(s):

1. NEI 99-01 Emergency Director Judgment Fuel Clad Potential Loss 6.A

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

RCB1

RPV water level **cannot** be restored and maintained > -162 in. (TAF) or **cannot** be determined

Definition(s):

None

Basis:

An RPV water level instrument reading of -162 in. indicates level is at the top of active fuel (TAF) (ref. 1). TAF is significantly lower than the normal operating RPV level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Containment barriers, and initiation of all ECCS. If RPV water level cannot be maintained above TAF, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the lowering level trend. The cause of the loss of RPV inventory is therefore assumed to be a LOCA. By definition, a LOCA event is a Loss of the RCS barrier.

When RPV water level cannot be determined, EOPs require entry to EOP-4, RPV Flooding. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained (ref. 2). The instructions in EOP-4 specify emergency depressurization of the RPV, which is defined to be a Loss of the RCS barrier (RCS Loss RCB3).

Note that EOP-1A, RPV Control, ATWS, may require intentionally lowering RPV water level to -162 in. and control level between the Minimum Steam Cooling RPV Water Level (MSCRWL) and the top of active fuel (ref. 3). Under these conditions, a high-power ATWS event exists and requires at least a Site Area Emergency classification in accordance with the System Malfunction - RPS Failure EALs.

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs ~~SA5~~ SA6 or ~~SS5~~ SS6 will dictate the need for emergency classification.

This water level corresponds to the top of active fuel and is used in the EOPs to indicate challenge to core cooling.

The RPV water level threshold is the same as Fuel Clad barrier Potential Loss threshold ~~2.A~~FCB2. Thus, this threshold indicates a Loss of the RCS barrier and Potential Loss of the Fuel Clad barrier and that appropriately escalates the emergency classification level to a Site Area Emergency.

This threshold is considered to be exceeded when, as specified in the site-specific EOPs, RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this RCS barrier Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

The term, "cannot be restored and maintained above," means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value cannot be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active fuel, but does not permit extended operation beyond the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

There is no RCS barrier Potential Loss threshold associated with RPV Water Level.

Reference(s):

1. EOP-1 RPV Control
2. EOP-4 RPV Flooding
3. EOP-1A RPV Control, ATWS
4. NEI 99-01 RPV Water Level RCS Loss 2.A

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

RCB2

UNISOLABLE break in **any** of the following:

- Main steam line
- RCIC steam line
- RWCU
- Feedwater

Definition(s):

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

Basis:

Failure to isolate the leak (from the Control Room or locally), within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

The conditions of this threshold include required containment isolation failures allowing a flow path to the environment. A release pathway outside primary containment exists when flow is not prevented by downstream isolations. In the case of a failure of both isolation valves to close but in which no downstream flowpath exists, emergency declaration under this threshold would not be required. Similarly, if the emergency response requires the normal process flow of a system outside containment (e.g., EOP requirement to bypass MSIV low RPV water level interlocks and maintain the main condenser as a heat sink using main turbine bypass valves), the threshold is not met. The combination of these threshold conditions represent the loss of both the RCS and Containment (see Loss CNB9) barriers and justifies declaration of a Site Area Emergency (i.e., Loss or Potential Loss of any two barriers).

Even though RWCU and Feedwater systems do not contain steam, they are included in the list because an UNISOLABLE break could result in the high-pressure discharge of fluid that is flashed to steam from relatively large volume systems directly connected to the RCS.

Large high-energy lines that rupture outside primary containment can discharge significant amounts of inventory and jeopardize the pressure-retaining capability of the RCS until they are isolated. If it is determined that the ruptured line cannot be promptly isolated, remotely or locally ~~from the Control Room~~, the RCS barrier Loss threshold is met.

Reference(s):

1. NEI 99-01 RCS Leak Rate RCS Loss 3.A

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

RCB3

Emergency Depressurization is required

Definition(s):

None

Basis:

Plant symptoms requiring Emergency RPV Depressurization per the EOPs are indicative of a loss of the RCS barrier (ref. 1, 2).

Emergency RPV Depressurization in accordance with the EOPs is indicative of a loss of the RCS barrier. If Emergency RPV Depressurization is performed, the plant operators are directed to open safety relief valves (SRVs)~~ and keep them open~~. Even though the RCS is being vented into the suppression pool, a Loss of the RCS barrier exists due to the diminished effectiveness of the RCS to retain fission products within its boundary.

EOP-1 Emergency Depressurization allows terminating the depressurization if necessary to maintain RCIC as an injection source. This would require closing the SRVs. Even though the SRVs may be reclosed, this threshold is still met due to the requirement for an Emergency Depressurization having been met (ref. 2).

Reference(s):

1. EOP-1 RPV Control Emergency Depressurization
2. EP FAQ 2015-003
3. NEI 99-01 RCS Leak Rate RCS Loss 3.B

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

RCB4

UNISOLABLE primary system leakage that results in exceeding **EITHER**:

- One or more EOP-3 Max Normal area radiation operating value (Table F-2)
- One or more Isolation Temperature alarms (Table F-2)

Table F-2 Secondary Containment Operating Values		
Area Temperatures		
Parameter	Isolation Temperature	Max Safe
Main Steam Line Tunnel	173°F (P601-19A-A1/A3/B1/B3)	200°F
RHR Equipment Area 1 (A)	117°F (P601-20A-B4)	200°F
RHR Equipment Area 2 (B)	117°F (P601-20A-B4)	200°F
RCIC Equipment Area	182°F (P601-21A-B6)	200°F
RWCU Pump Room 1 (A) / 2 (B)	165°F (P680-1A-A2/B2)	200°F
Area Radiation Levels		
Parameter	Max Normal	Max Safe
HPCS Area (1212) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RHR Equipment Room A (1213) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RHR Equipment Room B (1214) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RHR Equipment Room C (1215) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
LPCS Equipment Room (1216) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
HPCS Penetration Area (1217) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
LPCS Penetration Area (1218) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RCIC Equipment Room (1219) Grid 2	1.20E+02 mR/hr	9.5E+03 mR/hr

Definition(s):

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

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

Basis:

Failure to isolate the leak (from the Control Room or locally), within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

The presence of elevated general area temperatures or radiation levels in the Secondary Containment may be indicative of UNISOLABLE primary system leakage outside the containment. The EOP-3 Max Normal and Isolation Temperature alarm setpoint values in Table F-2 define this RCS threshold because they are the maximum normal operating/ Technical Specification Isolation values and signify the onset of abnormal system operation. When parameters reach this level, equipment failure or mis-operation may be occurring. Elevated parameters may also adversely affect the ability to gain access to or operate equipment within the affected area. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-3, Secondary Containment and Radioactivity Release Control (ref. 1).

In general, multiple indications should be used to determine if a primary system is discharging outside Containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the Secondary Containment since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room FLOODING, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the Secondary Containment.

Potential loss of RCS based on primary system leakage outside the primary containment is determined from EOP temperature or radiation Max Normal Operating values in areas such as main steam line tunnel, RCIC, ~~HPCI,~~ etc., which indicate a direct path from the RCS to areas outside primary containment.

A Max Normal Operating value is the highest value of the identified parameter expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly. An Isolation Temperature value is indicative of an UNISOLABLE leak when temperatures do not begin to recover as a result of the isolation actions following the alarm and represents a Technical Specification limiting value.

The indicators reaching the threshold barriers and confirmed to be caused by RCS leakage from a primary system warrant an Alert classification. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a reduction in the steam or water being discharged through an unisolated break in the system.

Reference(s):

1. EOP-3 Secondary Containment and Radioactivity Release Control
2. NEI 99-01 RCS Leak Rate RCS Potential Loss 3.A

Barrier: Reactor Coolant System

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

RCB5
Drywell pressure > 1.68 psid due to RCS leakage

Definition(s):

None

Basis:

The drywell high pressure scram setpoint is an entry condition to EOP-1, RPV Control. A high Containment pressure of greater than 0.3 psig is an entry condition to EOP-2, Primary Containment Control (ref. 1, 2). Normal containment pressure control functions (e.g., operation of drywell and containment cooling, vent using containment vessel purge, etc.) are specified in EOP-2 in advance of less desirable but more effective functions (e.g., Emergency Depressurization, etc.).

In the design basis, containment pressures above the drywell high pressure scram setpoint are assumed to be the result of a high-energy release into the containment for which normal pressure control systems are inadequate or incapable of reversing the rising pressure trend. Pressures of this magnitude, however, can be caused by non-LOCA events such as a loss of drywell cooling or inability to control containment vent/purge (ref. 3).

The threshold phrase "…due to RCS leakage" focuses the barrier failure on the RCS instead of the non-LOCA malfunctions that may adversely affect containment pressure. Drywell pressure greater than 1.68 psid with corollary indications (e.g., drywell temperature, indications of loss of RCS inventory) should therefore be considered a Loss of the RCS barrier. Loss of drywell cooling that results in pressure greater than 1.68 psid should not be considered an RCS barrier Loss.

The ~~(site-specific value)~~1.68 psid ~~primary containment pressure~~value is the drywell high pressure setpoint which indicates a LOCA by automatically initiating the ECCS or equivalent makeup system.

There is no RCS barrier Potential Loss threshold associated with ~~Primary Containment Pressure~~CTMT Conditions.

Reference(s):

1. EOP-1 RPV Control
2. EOP-2 Primary Containment Control
3. USAR Section 6.2.1 Containment Functional Design
4. NEI 99-01 Primary Containment Pressure RCS Loss 1.A

Barrier: Reactor Coolant System

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CTMT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

RCB6
Drywell radiation (RMS-RE20) > 30 R/hr

Definition(s):

N/A

Basis:

Under post-LOCA conditions coaxial cables used on the drywell post-accident monitors (RMS-RE20A/B) are susceptible to Thermally Induced Currents (TIC). These currents may cause the drywell PAMs to read falsely high (~469 R/hr) on a rapid temperature rise and read falsely low on a rapid temperature drop. When accident temperature conditions stabilize indicated radiation dose rates would be more accurate. The duration of the spurious signal would last approximately 15 minutes. During the period of false readings operators should rely on other indications of RCS leakage including a rise in drywell temperature and pressure (RCB5).

The drywell radiation monitor reading (38 R/hr rounded to 30 R/hr for readability) corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold ~~4.A~~FCB3 since it indicates a loss of the RCS Barrier only (ref. 1).

There is no RCS barrier Potential Loss threshold associated with ~~Primary Containment~~CTMT Radiation/RCS Activity.

Reference(s):

1. Calculation G13.18.9.4-045 Containment Doses for Emergency Action Levels (EALs)
2. NEI 99-01 Primary Containment Radiation RCS Loss 4.A

Barrier: Reactor Coolant System

Category: D. CTMT Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

RCB7
Any condition in the opinion of the Emergency Director that indicates loss of the RCS barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is lost.

Reference(s):

- NEI 99-01 Emergency Director Judgment RCS Loss 6.A

Barrier: Reactor Coolant System

Category: F. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

RCB8
Any condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that may be used by the Emergency Director 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.

Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Potential Loss 6.A

Barrier: Containment

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

CNB1

SAP entry is required

Definition(s):

None

Basis:

EOPs specify entry to the SAPs when core cooling is severely challenged. These EOPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined (ref. 1, 2).

The EOP conditions requiring SAP entry represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

This threshold is also a Loss of the Fuel Clad barrier (Loss FCB1). Since SAP entry occurs after core uncovery has occurred a Loss of the RCS barrier exists (Loss RCB1). SAP entry, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

The Potential Loss threshold is identical to the Fuel Clad Loss RPV Water Level threshold ~~A.1~~FCB1. The Potential Loss requirement for entry into the ~~SAGs~~ SAPs indicates adequate core cooling cannot be assured and that core damage is possible. BWR EPGs/SAGs (RBS term SAPs) specify the conditions when the EPGs are exited and SAGs are entered. Entry into SAGs is a logical escalation in response to the inability to assure adequate core cooling.

PRA studies indicate that the condition of this Potential Loss threshold could be a core melt sequence which, if not corrected, could lead to RPV failure and ~~increased~~ greater potential for primary containment failure. In conjunction with the RPV water level Loss thresholds in the Fuel Clad and RCS barrier columns, this threshold results in the declaration of a General Emergency.

There is no Containment barrier Loss threshold associated with RPV Water Level.

Reference(s):

1. EOP-1 RPV Control
2. EOP-4 RPV Flooding
3. EP FAQ 2015-004
4. NEI 99-01 RPV Water Level PC Potential Loss 2.A

Barrier: Containment

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

CNB2

UNISOLABLE primary system leakage that results in exceeding **EITHER**:

- One or more EOP-3 Max Safe area radiation operating value (Table F-2)
- One or more EOP-3 Max Safe area temperature operating value (Table F-2)

Table F-2 Secondary Containment Operating Values		
Area Temperatures		
Parameter	Isolation Temperature	Max Safe
Main Steam Line Tunnel	173°F (P601-19A-A1/A3/B1/B3)	200°F
RHR Equipment Area 1 (A)	117°F (P601-20A-B4)	200°F
RHR Equipment Area 2 (B)	117°F (P601-20A-B4)	200°F
RCIC Equipment Area	182°F (P601-21A-B6)	200°F
RWCU Pump Room 1 (A) / 2 (B)	165°F (P680-1A-A2/B2)	200°F
Area Radiation Levels		
Parameter	Max Normal	Max Safe
HPCS Area (1212) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RHR Equipment Room A (1213) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RHR Equipment Room B (1214) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RHR Equipment Room C (1215) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
LPCS Equipment Room (1216) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
HPCS Penetration Area (1217) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
LPCS Penetration Area (1218) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RCIC Equipment Room (1219) Grid 2	1.20E+02 mR/hr	9.5E+03 mR/hr

Definition(s):

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

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

Basis:

Failure to isolate the leak (from the Control Room or locally), within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

The presence of elevated general area temperatures or radiation levels in the secondary containment may be indicative of UNISOLABLE primary system leakage outside the containment. The Max Safe conditions define this Containment barrier threshold because they are indicative of problems in the secondary containment that are spreading and pose a threat to achieving a safe plant shutdown. This threshold addresses problematic discharges outside containment that may not originate from a high-energy line break. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-3, Secondary Containment and Radioactivity Release Control (ref. 1).

In general, multiple indications should be used to determine if a primary system is discharging outside containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the secondary containment since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room FLOODING, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the secondary containment.

The Max Safe Operating Temperature and the Max Safe Operating Radiation Level are each the highest value of these parameters at which neither: (1) equipment necessary for the safe shutdown of the plant will fail, nor (2) personnel access necessary for the safe shutdown of the plant will be precluded. EOPs utilize these temperatures and radiation levels to establish conditions under which RPV depressurization is required.

There is no Containment barrier Potential Loss threshold associated with RCS Leak Rate.

Reference(s):

1. EOP-3 Secondary Containment and Radioactivity Release Control
2. NEI 99-01 RCS Leak Rate PC Loss 3.C

Barrier: Containment

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

CNB3
UNPLANNED rapid drop in containment pressure following containment pressure rise

Definition(s):

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

Basis:

Rapid UNPLANNED loss of ~~primary~~ containment pressure (i.e., not attributable to ~~drywell spray~~containment cooling or condensation effects) following an initial pressure ~~increase~~ rise indicates a loss of primary containment integrity. ~~Primary containment pressure should increase as a result of mass and energy release into the primary containment from a LOCA. Thus, primary containment pressure not increasing under these conditions indicates a loss of primary containment integrity.~~

~~These~~ This thresholds ~~rely~~ relies on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

Reference(s):

1. NEI 99-01 Primary Containment Conditions PC Loss 1.A

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

CNB4

Containment pressure response **not** consistent with LOCA conditions

Definition(s):

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

Basis:

~~Rapid UNPLANNED loss of primary containment pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure increase indicates a loss of primary containment integrity.~~ Primary containment pressure should ~~increase~~ rise as a result of mass and energy release into the primary containment from a LOCA. Thus, primary containment pressure not ~~increasing~~ rising under these conditions indicates a loss of primary containment integrity.

These thresholds rely on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

Reference(s):

1. USAR Table 6.2-7, Results of Containment Response Analysis
2. USAR Table 6.2-1, Containment Design Parameters
3. NEI 99-01 Primary Containment Conditions PC Loss 1.B

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

CNB5

Containment pressure > 15 psig

Definition(s):

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

Basis:

When the containment pressure exceeds the maximum allowable value (15 psig) (ref. 1), containment venting may be required even if offsite radioactivity release rate limits will be exceeded (ref. 2). This pressure is based on the containment design pressure as identified in the accident analysis. If this threshold is exceeded, a challenge to the containment structure has occurred because assumptions used in the accident analysis are no longer VALID and an unanalyzed condition exists. This constitutes a Potential Loss of the Containment barrier even if a containment breach has not occurred.

The threshold pressure is the primary containment internal design pressure. Structural acceptance testing demonstrates the capability of the primary containment to resist pressures greater than the internal design pressure. A pressure of this magnitude is greater than those expected to result from any design basis accident and, thus, represent a Potential Loss of the Containment barrier.

Reference(s):

1. USAR Table 6.2-1, Containment Design Parameters
2. EOP-2 Primary Containment Control
3. NEI 99-01, Primary Containment Conditions PC Potential Loss 1.A

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

CNB6
Drywell or containment hydrogen concentration > 4%

Definition(s):

None

Basis:

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 hydrogen concentration has exceeded the minimum necessary to support a hydrogen burn (4%). The Igniter System is designed to prevent hydrogen accumulation by locally burning hydrogen in a controlled manner as soon as the hydrogen enters the containment atmosphere and reaches the igniters. For high rates of hydrogen production, ignition occurs at the lowest concentration that can support ignition. Following ignition, hydrogen is consumed through formation of diffusion flames where the gas enters the containment, thus controlling hydrogen concentration at approximately 4% (ref. 1).

If hydrogen concentration reaches or exceeds the lower flammability limit, as defined in plant EOPs, in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside the containment, loss of the Containment barrier could occur.

Reference(s):

1. EPSTG*0002 Appendix B EOP and SAP Bases
2. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.B

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

CNB7

Parameters **cannot** be restored and maintained within the safe zone of the HCTL curve (EOP Figure 2)

Definition(s):

None

Basis:

The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool temperature from which Emergency RPV Depressurization will not raise:

- Suppression chamber temperature above the maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized,

  OR

- Suppression chamber pressure above Primary Containment Pressure Limit~~-A~~, while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

The HCTL is a function of RPV pressure, suppression pool temperature and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant and therefore, the inability to maintain plant parameters below the limit constitutes a potential loss of containment.

The term "**cannot** be restored and maintained above" means the parameter value(s) is not able to be brought within the specified limit. The determination requires an evaluation of system performance and availability in relation to the parameter value and trend. A threshold prescribing declaration when a threshold value cannot be restored and maintained within a specified limit does not require immediate action simply because the current value is outside the limit, but does not permit extended operation outside the limit; the threshold must be considered reached as soon as it is apparent that operation within the limit cannot be attained.

Reference(s):

1. EOP-2 Primary Containment Control
2. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.C

Barrier: Containment

Category: D. CTMT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: C. CTMT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

CNB8

Containment radiation (RMS-RE16) > 12,000 R/hr

Definition(s):

None

Basis:

In order to reach this Containment barrier Potential Loss threshold, a loss of the RCS barrier (Loss RCB6) and a loss of the Fuel Clad barrier (Loss FCB3) have already occurred. This threshold, therefore, represents a General Emergency classification.

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that 20% of the fuel cladding has failed (ref. 1). 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 to a General Emergency.

There is no Containment barrier Loss threshold associated with CTMT Radiation/RCS Activity.

Reference(s):

1. Calculation G13.18.9.4-045 Containment Doses for Emergency Action Levels (EALs)
2. NEI 99-01 NEI 99-01 Primary Containment Radiation Potential Loss 4.A

Barrier: Containment

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

CNB9

UNISOLABLE direct downstream pathway to the environment exists after Containment isolation signal

Definition(s):

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

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

Basis:

Failure to isolate the leak (from the Control Room or locally), within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway directly to the environment. The concern is the UNISOLABLE open pathway to the environment. A failure of the ability to isolate any one line indicates a breach of containment integrity.

This threshold also applies to a containment bypass due to a HPCS or LPCS line break outside containment with injection check valve failure allowing an UNISOLABLE direct pathway for RCS release to the environment.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems or minor release pathways, such as instrument lines, not protected by the Primary Containment Isolation System (PCIS). Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include UNISOLABLE main steam line or RCIC steam line breaks, UNISOLABLE RWCU system breaks, and UNISOLABLE containment atmosphere vent paths. If the main condenser is available with an UNISOLABLE main steam line, there may be releases through the steam jet air ejectors and gland seal exhausters. These pathways are monitored, however, and do not meet the intent of an UNISOLABLE release path to the environment. These minor releases are assessed using the Category A, Abnormal Rad Levels / Rad Effluent, EALs.

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.

EOP-2 Primary Containment Control, may specify primary containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1). Under these conditions with a VALID containment isolation signal, the Containment barrier should be considered lost. Refer to CNB10.

Following the leakage of RCS mass into primary containment and a rise in primary containment pressure, there may be minor radiological releases associated with allowable primary containment leakage through various penetrations or system components. Minor releases may also occur if a primary containment isolation valve(s) fails to close but the primary containment atmosphere escapes to an enclosed system. These releases do not constitute a loss or potential loss of primary containment but should be evaluated using the Recognition Category A ICs.

There is no Containment barrier Potential Loss threshold associated with CTMT Integrity or Bypass.

Reference(s):

1. EOP-2 Primary Containment Control
2. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.A

Barrier: Containment

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

CNB10 Intentional Containment venting per EOPs

Definition(s):

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

Basis:

EOP-2, Primary Containment Control, may specify containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded. Under these conditions, with a VALID primary containment isolation signal, the threshold is met when the operator begins venting the containment in accordance with Enclosure 21, not when actions are taken to bypass interlocks prior to opening the vent valves (ref. 1).

~~EOPs may direct primary containment isolation valve logic(s) to be intentionally bypassed, even if offsite radioactivity release rate limits will be exceeded. Under these conditions with a valid primary containment isolation signal, the containment should also be considered lost if primary containment venting is actually performed.~~

Intentional venting of primary containment for primary containment pressure or combustible gas control to the secondary containment and/or the environment is a Loss of the Containment. Venting for primary containment pressure control when not in an accident situation (e.g., to control pressure below the drywell high pressure scram setpoint) does not meet the threshold condition.

There is no Containment barrier Potential Loss threshold associated with CTMT Integrity or Bypass.

Reference(s):

1. EOP-2 Primary Containment Control
2. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.B

Barrier: Containment

Category: E. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

CNB11
Any condition in the opinion of the Emergency Director that indicates loss of the Containment barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is lost.

Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

Barrier: Containment

Category: E. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

CNB12 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment 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.

Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Potential Loss 6.A

Category H – Hazards and Other Conditions Affecting Plant Safety

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

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

1. Security

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

2. Seismic Event

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

3. Natural or Technological Hazard

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

4. Fire

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

5. Hazardous Gas

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

6. Control Room Evacuation

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

7. 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 which may warrant classification and subsequent implementation of the Emergency Plan, a provision for classification of emergencies based on operator/management experience and judgment is still necessary. The EALs of this category provide the Emergency Director the latitude to classify emergency conditions consistent with the established classification criteria based upon Emergency Director judgment.

Category: H – Hazards

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 RBS Security Shift Supervision

OR

Notification of a credible security threat directed at the site

OR

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

Mode Applicability:

All

Definition(s):

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

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

OWNER CONTROLLED AREA (OCA) - For the purposes of classification, the Security Owner Controlled Area (SOCA) or the area between the SOCA Fence and the PROTECTED AREA Boundary.

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

PROTECTED AREA - The area within the perimeter of the RBS security fence.

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

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

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

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

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

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

Basis:

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

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

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

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

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

The third threshold ~~EAL #3~~ addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with AOP-0063 Outside Threats (ref. 2)~~(site-specific procedure)~~.

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

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

Reference(s):

1. RBS Security Plan
2. AOP-0063 Outside Threats
3. AOP-0054 Security Events
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HA1.1 Alert

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

OR

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

Mode Applicability:

All

Definition(s):

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

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

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

OWNER CONTROLLED AREA (OCA) - For the purposes of classification, the Security Owner Controlled Area (SOCA) or the area between the SOCA Fence and the PROTECTED AREA Boundary.

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

PROTECTED AREA - The area within the perimeter of the RBS security fence.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the SECURITY 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.

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

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

The second threshold ~~EAL #2~~ addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and OROs are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with AOP-0063 Outside Threats (ref. 2)~~(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 SECURITY OWNER CONTROLLED AREA was intentional (i.e., a HOSTILE ACTION). It is expected, although not certain, that notification by an appropriate Federal agency to the site would clarify this point. In this case, the appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. The emergency declaration, including one based on other ICs/EALs, should not be unduly delayed while awaiting notification by a Federal agency.

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

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

Reference(s):

1. RBS Security Plan
2. AOP-0063 Outside Threats
3. AOP-0054 Security Events
4. NEI 99-01 HA1

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 RBS Security Shift Supervision	

Mode Applicability:

All

Definition(s):

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

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

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

OWNER CONTROLLED AREA (OCA) - For the purposes of classification, the Security Owner Controlled Area (SOCA) or the area between the SOCA Fence and the PROTECTED AREA Boundary.

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

PROTECTED AREA - The area within the perimeter of the RBS security fence.

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. 1, 2, 3).

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

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Site Area Emergency declaration will mobilize 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~~ EAL does not apply to ~~a HOSTILE ACTION directed at an ISFSI Protected Area located outside the PROTECTED AREA; such an attack should be assessed using IC HA1. It also does not apply to~~ incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

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

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

Reference(s):

1. RBS Security Plan
2. AOP-0063 Outside Threats
3. AOP-0054 Security Events
4. NEI 99-01 HS1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE levels

EAL:

HU2.1 Unusual Event

Seismic event > OBE as indicated by **EITHER** of the following:

- Annunciator P680-02A-C06, SEISMIC EVENT HIGH
- Annunciator P680-02A-B06, SEISMIC EVENT HIGH/HIGH and amber lights illuminated on H13-P869 ERS-NBI101

Mode Applicability:

All

Definition(s):

None

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 U. S. Geological Survey (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~~ EAL CA6.1 or ~~SA9~~SA8.1.

To avoid inappropriate emergency classification resulting from spurious actuation of the seismic instrumentation or felt motion not attributable to seismic activity, an offsite agency (USGS, National Earthquake Information Center (NEIC)) can confirm that an earthquake has occurred in the area of the plant. Such confirmation should not, however, preclude a timely emergency declaration based on receipt of the OBE alarm. If requested, provide the analyst with the following RBS coordinates: **30° 45' 26" north latitude, 91° 19' 54" west longitude** (ref. 3). Alternatively, near real-time seismic activity can be accessed via the NEIC website.

Reference(s):

1. ARP-680-02 P680-02 Alarm Response
2. AOP-0028 Seismic Event
3. USAR section 2.1.1.1 Specification of Location
4. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

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

Mode Applicability:

All

Definition(s):

PROTECTED AREA - The area within the perimeter of the RBS security fence.

Basis:

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

~~EAL #1~~ This EAL 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, F, S or C.

If damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under EAL CA6.1 or SA8.1.

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

Reference(s):

1. AOP-0029 Severe Weather Operation
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.2 Unusual Event

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

Mode Applicability:

All

Definition(s):

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

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

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

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

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

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

Basis:

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

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

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

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

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

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

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

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

Refer to EAL CA6.1 or SA8.1 for internal FLOODING affecting one or more SAFETY SYSTEM trains.

Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.3 Unusual Event

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

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - The area within the perimeter of the RBS security fence.

Basis:

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

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

~~This EAL addresses flooding of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, 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 a~~n offsite~~ location outside the PROTECTED AREA and of sufficient magnitude to IMPEDE the movement of personnel within the PROTECTED AREA.

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

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

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

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

Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.4	**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

Definition(s):

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

Basis:

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

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

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

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

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

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

Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.1 Unusual Event

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

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

AND

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table H-1 Fire Areas
• Reactor Building • Auxiliary Building • Fuel Building • Control Building • Standby Cooling Tower • Diesel Generator Building • Tunnels (B, D,E, F, G)

Mode Applicability:

All

Definition(s):

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

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

Basis:

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

~~EAL #1~~

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

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report. ~~EAL #2~~

~~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~~ EAL CA6.1 or ~~SA9~~SA8.1.

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

Table H-1 Fire Areas are those areas that contain equipment necessary for safe operation and shutdown of the plant (ref. 1).

Reference(s):

1. AOP-0052 Fire Outside the Main Control Room in Areas Containing Safety Related Equipment

2. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.2 Unusual Event

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

AND

The fire alarm is 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table H-1 Fire Areas
• Reactor Building • Auxiliary Building • Fuel Building • Control Building • Standby Cooling Tower • Diesel Generator Building • Tunnels (B, D,E, F, G)

Mode Applicability:

All

Definition(s):

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

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

Basis:

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

~~EAL #1~~

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

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

~~EAL #2~~

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

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via ~~IC~~ EAL CA6.1 or ~~SA9~~SA8.1.

The 30 minute requirement begins upon receipt of a single VALID fire detection system alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field. Actual field reports must be made within the 30 minute time limit or a classification must be made. If a fire is verified to be occurring by field report, classification shall be made based on EAL HU4.1, with the 15 minute requirement beginning with the verification of the fire by field report.

Table H-1 Fire Areas are those areas that contain equipment necessary for safe operation and shutdown of the plant (ref. 1).

Reference(s):

1. AOP-0052 Fire Outside the Main Control Room in Areas Containing Safety Related Equipment
2. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.3 Unusual Event

A FIRE within the 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - The area within the perimeter of the RBS security fence.

Basis:

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

~~EAL #1~~

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

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

~~EAL #2~~

~~This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the~~

~~30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.~~

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

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

~~EAL #3~~

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

~~This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA. [*Sentence for plants with an ISFSI outside the plant Protected Area*]EAL #4~~

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

~~Basis-Related Requirements from Appendix R~~

~~Appendix R to 10 CFR 50, states in part:~~

> ~~Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."~~
>
> ~~When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.~~
>
> ~~Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to~~

~~limit fire damage to those systems required to mitigate the consequences of design basis accidents.~~

~~In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.~~

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via ~~IC~~ EAL CA6.1 or ~~SA9~~SA8.1.

Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.4 Unusual Event

A FIRE within the PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - The area within the perimeter of the RBS security fence.

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~~ EAL CA6.1 or ~~SA9~~SA8.1.

Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gas

Initiating Condition: Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

HA5.1 Alert

Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 room or area

AND

Entry into the room or area is prohibited or IMPEDED (Note 5)

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

Table H-2 Safe Operation & Shutdown Rooms/Areas

Room/Area	Mode
Auxiliary Building 70' RHR B Pump Room	3
Auxiliary Building 80' RHR A Pump Room	3
Auxiliary Building 114' West	3
Control Building 95' Div 1 RSS Room	3

Mode Applicability:

3 – Hot Shutdown

Definition(s):

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

Basis:

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

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

Evaluation of the IC and EAL does not require atmospheric sampling; it only requires the Emergency Director's judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly IMPEDE procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as IMPEDED if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

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

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode ~~4~~3.
- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).
- ~~The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).~~
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or IMPEDE a required action.
- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

This EAL does not apply to firefighting activities that generate smoke and that automatically or manually activate a fire suppression system in an area~~, or to intentional inerting of containment. (BWR only)~~.

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

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).

EAL HA5.1 mode applicability has been limited to the mode limitations of Table H-2 (Mode 3 only).

Reference(s):

1. Attachment 2 Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases
2. NEI 99-01 HA5

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

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

EAL:

HA6.1 Alert

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

Transfer of plant control begins when the last licensed operator leaves the Control Room.

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

Reference(s):

1. AOP-0031 Shutdown from Outside the Main Control Room
2. NEI 99-01 HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

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

EAL:

HS6.1 Site Area Emergency

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels

AND

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

- Reactivity (Modes 1 and 2 **only**)
- RPV water level
- RCS heat removal

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refueling

Definition(s):

None

Basis:

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

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

Transfer of plant control and the time period to establish control begins when the last licensed operator leaves the Control Room.

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

Reference(s):

1. AOP-0031 Shutdown from Outside the Main Control Room
2. EP FAQ 2015-014
3. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist that 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

Definition(s):

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

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

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

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

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

Basis:

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

Reference(s):

1. NEI 99-01 HU7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist that 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

Definition(s):

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

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

OWNER CONTROLLED AREA (OCA) - For the purposes of classification, the Security Owner Controlled Area (SOCA) or the area between the SOCA Fence and the PROTECTED AREA Boundary.

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

PROTECTED AREA - The area within the perimeter of the RBS security fence.

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.

Reference(s):

1. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist that 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

Definition(s):

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

OWNER CONTROLLED AREA (OCA) - For the purposes of classification, the Security Owner Controlled Area (SOCA) or the area between the SOCA Fence and the PROTECTED AREA Boundary.

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

PROTECTED AREA - The area within the perimeter of the RBS security fence.

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline.

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.

Reference(s):

1. NEI 99-01 HS7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist that 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

Definition(s):

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

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

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

OWNER CONTROLLED AREA (OCA) - For the purposes of classification, the Security Owner Controlled Area (SOCA) or the area between the SOCA Fence and the PROTECTED AREA Boundary.

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

PROTECTED AREA - The area within the perimeter of the RBS security fence.

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.

Reference(s):

1. NEI 99-01 HG7

Category S – System Malfunction

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

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

The events of this category pertain to the following subcategories:

1. Loss of Emergency AC Power

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

2. Loss of Vital DC Power

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

3. Loss of Control Room Indications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant rise 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 pressure 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 scrams. In the plant licensing basis, postulated failures of the RPS to complete a reactor scram comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean any scram failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and containment integrity.

7. Loss of Communications

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

8. Hazardous Event Affecting Safety Systems

Various natural and technological events that result in degraded plant SAFETY SYSTEM performance or significant VISIBLE DAMAGE warrant emergency classification under this subcategory.

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SU1.1 Unusual Event

Loss of **all** offsite AC power capability, Table S-1, to DIV I and DIV II 4.16 KV ENS buses for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-1 AC Power Sources
Offsite • 1RTX-XSR1C • 1RTX-XSR1D
Onsite • EGS-EG1A • EGS-EG1B

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

The DIV III bus (1E22*S004) is not credited because it only supplies power to the HPCS pump and associated loads, not any long term decay heat removal systems. In particular, suppression pool cooling mechanisms would be essential subsequent to a station blackout.

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC ~~emergency~~ ENS buses. This condition represents a potential reduction in the level of safety of the plant.

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the ~~emergency~~ ENS 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.

Reference(s):

1. USAR Section 8.1 Electric Power Introduction
2. EE-001AC Startup Electrical Distribution Chart
3. SOP-0046 4.16 KV System
4. AOP-0004 Loss of Offsite Power
5. NEI 99-01 SU1

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SA1.1 Alert

AC power capability, Table S-1, to DIV I and DIV II 4.16 KV ENS buses 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-1 AC Power Sources

Offsite

- 1RTX-XSR1C
- 1RTX-XSR1D

Onsite

- EGS-EG1A
- EGS-EG1B

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

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

Basis:

The DIV III bus (1E22*S004) is not credited because it only supplies power to the HPCS pump and associated loads, not any long term decay heat removal systems. In particular, suppression pool cooling mechanisms would be essential subsequent to a station blackout.

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an ~~emergency~~ ENS 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 ENS emergency power sources (e.g., onsite diesel generators) with a single train of ~~emergency~~ ENS buses being back-fed from the unit main generator.
- A loss of ENS emergency power sources (e.g., onsite diesel generators) with a single train of ENS 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.

This EAL is the hot condition equivalent of~~to~~ the cold condition EAL CU2.1.

Reference(s):

1. USAR Section 8.1 Electric Power Introduction
2. EE-001AC Startup Electrical Distribution Chart
3. SOP-0046 4.16 KV System
4. AOP-0004 Loss of Offsite Power
5. NEI 99-01 SA1

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SS1.1 Site Area Emergency

Loss of **all** offsite and **all** onsite AC power capability to DIV I and DIV II 4.16 KV ENS buses for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

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

Basis:

This IC addresses a 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. Mitigative strategies using other power sources (Division III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout. 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, FG1 or SG1.

This EAL is the hot condition equivalent of the cold condition EAL CA2.1.

Reference(s):

1. USAR Section 8.1 Electric Power Introduction
2. EE-001AC Startup Electrical Distribution Chart
3. SOP-0046 4.16 KV System
4. AOP-0004 Loss of Offsite Power
5. NEI 99-01 SS1

Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to ENS buses

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power capability to DIV I and DIV II 4.16 KV ENS buses

AND EITHER:

- Restoration of at least one 4.16 KV ENS bus in < 4 hours is **not** likely (Note 1)
- RPV water level **cannot** be restored and maintained > -187 in.

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

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

Basis:

Indication of continuing core cooling degradation is manifested by the inability to restore and maintain RPV water level above the Minimum Steam Cooling Reactor Water Level (-187 in.) (ref. 5). Core submergence is the most desirable means of core cooling, however when RPV level is below TAF, the uncovered portion of the core can be cooled by less reliable means (i.e., steam cooling or spray cooling).

This IC addresses a prolonged loss of all power sources to AC ENS 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. Mitigative strategies using other power sources (Division III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout. 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 ENS 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 a~~n~~ ~~increased~~ greater likelihood of challenges to multiple fission product barriers. 4 hours is the site-specific SBO coping analysis time (ref. 6).

The estimate for restoring at least one ENS 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.

Reference(s):

1. USAR Section 8.1 Electric Power Introduction
2. EE-001AC Startup Electrical Distribution Chart
3. SOP-0046 4.16 KV System
4. AOP-0004 Loss of Offsite Power
5. EOP-1 RPV Control
6. USAR Appendix 15C Station Blackout
7. NEI 99-01 SG1

Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** ENS AC and vital DC power sources for 15 minutes or longer

EAL:

SG1.2 General Emergency

Loss of **all** offsite and **all** onsite AC power capability to DIV I and DIV II 4.16 KV ENS buses for ≥ 15 min. (Note 1)

AND

Indicated voltage is < 105 VDC on Safety Related DIV I and DIV II 125 VDC buses for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

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

Basis:

Safety Related DC buses ENB-SWG01A (DIV I) and ENB-SWG01B (DIV II) feed the Division I and Division II loads respectively. The Division I and Division II batteries each have 60 cells with a specific minimum voltage of 1.75 volts/cell. These cell voltages yield minimum design bus voltages of 105 VDC (ref. 5).

This IC addresses a concurrent and prolonged loss of both emergency ENS AC and Vital DC power. A loss of all emergency ENS 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. Mitigative strategies using other power sources (Division III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both emergency ENS AC and Vital DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

Reference(s):

1. USAR Section 8.1 Electric Power Introduction
2. EE-001AC Startup Electrical Distribution Chart
3. SOP-0046 4.16 KV System
4. AOP-0004 Loss of Offsite Power
5. Safety Related Battery Specification 244.521
6. USAR 8.3.2 DC Power Systems
7. NEI 99-01 SG8

Category: S – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

Initiating Condition: Loss of **all** vital DC power for 15 minutes or longer

EAL:

SS2.1	**Site Area Emergency**
Indicated voltage is < 105 VDC on Safety Related DIV I and DIV II 125 VDC buses for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

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

Basis:

Safety Related DC buses ENB-SWG01A (DIV I) and ENB-SWG01B (DIV II) feed the Division I and Division II loads respectively. The Division I and Division II batteries each have 60 cells with a specific minimum voltage of 1.75 volts/cell. These cell voltages yield minimum design bus voltages of 105 VDC (ref. 1).

This IC addresses a loss of vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs AG1, FG1 or SG1~~SG8~~.

This EAL is the hot condition equivalent of the cold condition EAL CU4.1.

Reference(s):

1. Safety Related Battery Specification 244.521
2. USAR 8.3.2 DC Power Systems
3. 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

EAL:

SU3.1 Unusual Event

An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-2 Safety System Parameters
• Reactor power • RPV water level • RPV pressure • Containment pressure • Suppression Pool water level • Suppression Pool temperature

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

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

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

Basis:

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 water level ~~[*BWR*]~~ and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for ~~reactor vessel level [PWR] /~~ RPV water level ~~[BWR]~~ cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via ~~IC~~ EAL SA3.1~~SA2~~.

Reference(s):

1. USAR 7.5 Safety-Related Display Instrumentation
2. NEI 99-01 SU2

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress

EAL:

SA3.1 Alert

An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

Any significant transient is in progress, Table S-3

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-2 Safety System Parameters
• Reactor power
• RPV water level
• RPV pressure
• Containment pressure
• Suppression Pool water level
• Suppression Pool temperature

Table S-3 Significant Transients
• Reactor scram
• Runback > 25% thermal reactor power
• Electrical load rejection > 25% full electrical load
• ECCS injection
• Thermal power oscillations > 10%

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

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

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

Basis:

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, RPV water level and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for ~~reactor vessel level [*PWR*] /~~ RPV water level ~~[*BWR*]~~ cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via IC~~s~~ FS1 or ~~IC~~ AS1

Reference(s):

1. USAR 7.5 Safety-Related Display Instrumentation
2. NEI 99-01 SA2

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

SU4.1	**Unusual Event**
Offgas Pretreatment radiation monitor high alarm (P601-22A-F03, OFF GAS PRE-TREAT HIGH RADIATION)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

The Offgas Pretreatment monitors radioactivity in the Offgas system downstream of the Offgas condenser. The monitor detects the radiation level that is attributable to the fission gases produced in the reactor and transported with steam through the turbine to the condenser. The High alarm indicates that the radioactivity present at the recombiner effluent discharge is approaching the Technical Specification 3.7.4 limit. The nominal setpoint of 1.5 times the full power process background radiation level ensures that the activity will not exceed a value corresponding to the Technical Specification LCO 3.7.4 allowable release rate. (ref. 1)

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 ICs.

Reference(s):

1. TRM section 3.3.7.8.2 Offgas System Radiation Monitoring Instrumentation
2. USAR 11.5 Process and Effluent Radiological Monitoring and Sampling Systems
3. Technical Specification 3.7.4 Main Condenser Offgas
4. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

SU4.2 Unusual Event

Coolant activity > 0.2 μCi/gm dose equivalent I-131 for > 48 hours

OR

Coolant activity > 4.0 μCi/gm dose equivalent I-131 instantaneous

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category A ICs.

Reference(s):

1. Technical Specification B 3.4.8, RCS Specific Activity bases
2. USAR Section 15.6.4 Steam System Piping Break Outside Containment
3. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1 Unusual Event

RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. (Note 1)

OR

RCS identified leakage > 25 gpm for ≥ 15 min. (Note 1)

OR

Leakage from the RCS to a location outside Containment > 25 gpm for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

Basis:

Failure to isolate the leak (from the Control Room or locally) within 15 minutes, or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

Identified leakage is leakage into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a collecting sump; or leakage into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage.

Unidentified leakage is all leakage into the drywell that is not identified leakage (ref. 2, 3).

Pressure boundary leakage is leakage through a non-isolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall (ref. 2, 3).

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

The first and second EAL conditions ~~EAL #1 and EAL #2~~ are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications). The third condition ~~EAL #3~~ addresses an RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These conditions ~~EALs~~ thus apply to leakage into the containment, ~~a secondary-side system (e.g., steam generator tube leakage in a PWR)~~ or a location outside of containment.

The leak rate values for each condition ~~EAL~~ were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). The first condition ~~EAL #1~~ uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

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, 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~~ 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 or F.

Reference(s):

1. USAR Section 5.2.5 Reactor Coolant Pressure Boundary and ECCS Leakage Detection System
2. Technical Specification Definitions Section 1.1
3. Technical Specification 3.4.5
2. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor

EAL:

SU6.1 Unusual Event

An automatic scram did **not** shut down the reactor as indicated by reactor power > 5% after **any** RPS setpoint is exceeded

AND

A subsequent automatic scram or manual scram action taken at the reactor control console (Mode Switch, Manual PBs, ARI) is successful in shutting down the reactor as indicated by reactor power ≤ 5% (APRM downscale)

Note 8: A manual scram 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, 2 - Startup

Definition(s):

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

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(trip [*PWR*] /~~ scram ~~[*BWR*])~~ that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic ~~(trip [*PWR*] /~~ scram ~~[*BWR*])~~ is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function. A reactor scram is automatically initiated by the RPS when certain continuously monitored parameters exceed predetermined setpoints (ref. 1).

A successful scram has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power to or below the APRM downscale setpoint of 5% (ref. 4).

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., mode switch, manual scram pushbuttons, or ARI initiation). Reactor shutdown achieved by use of alternate control rod insertion methods (i.e., EOP-1A Enclosure 26) does not constitute a successful manual scram (ref. 2).

Following any automatic RPS scram signal, operating procedures (e.g., EOP-1A) prescribe insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved. Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Unusual Event (ref. 3).

Taking the Mode Switch to Shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

For the purposes of this EAL, a successful automatic initiation of ARI that reduces reactor power to < 5% is not considered a successful automatic scram. If automatic initiation of ARI has occurred and caused reactor shutdown, the automatic RPS scram must have failed. ARI is a backup means of inserting control rods in the unlikely event that an automatic RPS scram signal exists but the reactor continues to generate significant power. However, a successful automatic or manual initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

In the event that the operator identifies a reactor scram is IMMINENT and initiates a successful manual reactor scram before the automatic scram setpoint is reached, no declaration is required. The successful manual scram of the reactor before it reaches its automatic scram setpoint or reactor scram signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. If manual reactor scram 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 scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

Following the failure on an automatic reactor ~~(trip[PWR] /~~ scram ~~[BWR])~~, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor ~~(trip[PWR] /~~ scram ~~[BWR]~~)). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor ~~(trip [PWR] /~~ scram ~~[BWR])~~ is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor ~~(trip[PWR] /~~ scram ~~[BWR])~~) using a different switch).

Depending upon several factors, the initial or subsequent effort to manually ~~(trip [PWR] /~~scram ~~[BWR])~~ the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor ~~(trip [PWR] /~~scram ~~[BWR])~~ signal. If a subsequent manual or automatic ~~(trip [PWR] /~~ scram ~~[BWR])~~ is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor ~~(trip[PWR] /~~scram ~~[BWR])~~). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles". Taking the Reactor Mode Switch to Shutdown is considered to be a manual scram action. ~~[BWR]~~

The plant response to the failure of an automatic or manual reactor ~~(trip [PWR] /~~scram ~~[BWR])~~ will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via ~~IC~~ EAL SA6.1~~SA5~~. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6~~SA5~~ 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 generated as a result of plant work causes a plant transient that results in a condition 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~~ associated EALs are applicable, and should be evaluated.
- If the signal generated as a result of plant work does not cause a plant transient and the scram failure is determined through other means (e.g., assessment of test results), then this IC and ~~the~~ associated EALs are not applicable and no classification is warranted.

Reference(s):

1. Technical Specification Table 3.3.1.1-1 Reactor Protection System Instrumentation
2. EOP-1A RPV Control, ATWS
3. EOP-1 RPV Control
4. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor

EAL:

SU6.2 Unusual Event

A manual scram did **not** shut down the reactor as indicated by reactor power > 5% after **any** manual scram action was initiated

AND

A subsequent automatic scram or manual scram action taken at the reactor control console (Mode Switch, Manual PBs, ARI) is successful in shutting down the reactor as indicated by reactor power ≤ 5% (APRM downscale) (Note 8)

Note 8: A manual scram 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, 2 - Startup

Definition(s):

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

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

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

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

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

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(trip [*PWR*] /~~ scram ~~[*BWR*])~~ that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic ~~(trip [*PWR*] /~~ scram ~~[*BWR*])~~ is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

This EAL addresses a failure of a manually initiated scram in the absence of having exceeded an automatic RPS trip setpoint and a subsequent automatic or manual scram is successful in shutting down the reactor (reactor power ≤ 5%) (ref. 1).

A successful scram has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power to or below the APRM downscale setpoint of 5%.

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., Mode Switch, manual scram pushbuttons, or ARI initiation). Reactor shutdown achieved by use of alternate control rod insertion methods (i.e., EOP-1A Enclosure 26) does not constitute a successful manual scram (ref. 2).

Taking the Mode Switch to Shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

Successful automatic or manual initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

If both subsequent automatic and subsequent manual reactor scram 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 scram, the event escalates to an Alert under EAL SA6.1.

Following the failure on an automatic reactor ~~(trip [PWR] /~~ scram ~~[BWR])~~, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor ~~(trip [PWR] /~~ scram ~~[BWR])~~). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor ~~(trip [PWR] /~~ scram ~~[BWR])~~ is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor ~~(trip [PWR] /~~ scram ~~[BWR])~~) using a different switch~~)~~. Depending upon several factors, the initial or subsequent effort to manually ~~(trip [PWR] /~~ scram ~~[BWR])~~ the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor ~~(trip [PWR] /~~ scram ~~[BWR])~~ signal. If a subsequent manual or automatic ~~(trip [PWR] /~~ scram ~~[BWR])~~ is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor ~~(trip [PWR] /~~ scram ~~[BWR])~~). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

Taking the Reactor Mode Switch to Shutdown is considered to be a manual scram action. ~~[BWR]~~

The plant response to the failure of an automatic or manual reactor ~~(trip [PWR] /~~ scram ~~[BWR])~~ will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC ~~SA5~~SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC ~~SA5~~ SA6 or FA1, an Unusual Event declaration is appropriate for this event.

~~A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.~~

Should a reactor ~~(trip [PWR] /~~ scram ~~[BWR])~~ signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal generated as a result of plant work causes a plant transient that that results in a condition 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~~ associated EALs are applicable, and should be evaluated.
- If the signal generated as a result of plant work 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~~ associated EALs are not applicable and no classification is warranted.

Reference(s):

1. Technical Specification Table 3.3.1.1-1 Reactor Protection System Instrumentation
2. EOP-1A RPV Control, ATWS
3. EOP-1 RPV Control
4. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are **not** successful in shutting down the reactor

EAL:

SA6.1 Alert

An automatic or manual scram fails to shut down the reactor as indicated by reactor power > 5%

AND

Manual scram actions taken at the reactor control console (Mode Switch, Manual PBs, ARI) are **not** successful in shutting down the reactor as indicated by reactor power > 5% (Note 8)

Note 8: A manual scram 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, 2 - Startup

Definition(s):

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

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

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

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

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

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(trip [*PWR*] /~~ scram ~~[*BWR*])~~ that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

This EAL addresses any automatic or manual reactor scram signal that fails to shut down the reactor followed by a subsequent manual scram 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 (> 5%).

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., Mode Switch, manual scram pushbuttons, or ARI initiation). Reactor shutdown achieved by use of alternate control rod insertion methods (i.e., EOP-1A Enclosure 26) does not constitute a successful manual scram (ref. 2).

For the purposes of this EAL, a successful automatic initiation of ARI that reduces reactor power to or below 5% is not considered a successful automatic scram. If automatic actuation of ARI has occurred and caused reactor shutdown, the automatic RPS scram must have failed. ARI is a backup means of inserting control rods in the unlikely event that an automatic RPS scram signal exists but the reactor continues to generate significant power. However, a successful automatic initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor (~~trip [PWR] /~~ scram ~~[BWR]~~)). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control consoles (e.g., locally opening breakers). Actions taken at back panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

Taking the Reactor Mode Switch to Shutdown is considered to be a manual scram action. ~~[BWR]~~

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 shutdown the reactor is prolonged enough to cause a challenge to ~~the core cooling [PWR] /~~ RPV water level ~~[BWR]~~ or RCS ~~RCS~~ heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC ~~SS5~~SS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC ~~SS5~~ 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.~~

Reference(s):

1. Technical Specification Table 3.3.1.1-1 Reactor Protection System Instrumentation
2. EOP-1A RPV Control, ATWS
3. EOP-1 RPV Control
4. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to RPV water level or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual scram 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:

RPV water level **cannot** be restored and maintained > -187 in.

OR

Heat Capacity Temperature Limit (HCTL) exceeded (EOP Figure 2)

Mode Applicability:

1 - Power Operation, 2 - Startup

Definition(s):

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

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

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

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

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

Basis:

This IC addresses a 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.

This EAL addresses the following:

- Any automatic reactor scram signal followed by a manual scram that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the SAFETY SYSTEMS were designed (EAL SA6.1), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

Reactor shutdown achieved by use of control rod insertion methods in EOP-1A Enclosure 26 are also credited as a successful shutdown provided reactor power can be reduced to or below the APRM downscale trip setpoint before indications of an extreme challenge to either core cooling or heat removal exist. (ref. 1)

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.

Indication that core cooling is extremely challenged is manifested by inability to restore and maintain RPV water level above the Minimum Steam Cooling RPV Water Level (MSCRWL) (ref. 1). The MSCRWL is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500°F. This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core spray cooling could be a precursor of a core melt sequence (ref. 2).

The Heat Capacity Temperature Limit (HCTL, EOP Figure 2) is the highest suppression pool water temperature from which Emergency RPV Depressurization will not raise suppression pool temperature above the maximum design suppression pool temperature.

The HCTL is a function of RPV pressure and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant. This threshold is met when the final step of section SPT in EOP-2, Primary Containment Control, is reached (ref. 3). This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature.

In some instances, the emergency classification resulting from this ~~IC/~~EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shut down the reactor. The inclusion of this ~~IC and~~ EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

~~A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.~~

Escalation of the emergency classification level would be via IC AG1 or FG1.

Reference(s):

1. EOP-1A, RPV Control, ATWS
2. EOP-4, RPV Flooding
3. EOP-2, Primary Containment Control
4. NEI 99-01 SS5

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

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

EAL:

SU7.1 Unusual Event

Loss of **all** Table S-4 onsite communication methods

OR

Loss of **all** Table S-4 State and local agency communication methods

OR

Loss of **all** Table S-4 NRC communication methods

Table S-4 Communication Methods

System	Onsite	State/ Local	NRC
Plant radio system	X		
Plant Paging System	X		
Sound powered phones	X		
In-plant telephones	X		
Emergency Notification System (ENS)			X
Commercial Telephone System		X	X
Satellite Phones		X	X
State of Louisiana Radio		X	
State and Local Hotline radio		X	
INFORM Notification System		X	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

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

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

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

The second EAL condition ~~EAL #2~~ addresses a total loss of the communications methods used to notify all State and local agencies~~OROs~~ of an emergency declaration. The ~~OROs~~ State and local agencies referred to here are the Louisiana Department of Environmental Quality, Governor's Office of Homeland Security and Emergency Preparedness, Five Local Parishes Office of Homeland Security and Emergency Preparedness and 24 hour notification points, Mississippi Emergency Management Agency and the Mississippi Highway Patrol. ~~(see Developer Notes)~~

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

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

Reference(s):

1. RBS Emergency Plan Section 13.3.6.1.5.4 Communications
2. RBS Emergency Plan Section 13.3.6.2.1 Site Communications
3. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 8 – Hazardous Event Affecting Safety Systems

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

EAL:

SA8.1 Alert

The occurrence of **any** Table S-5 hazardous event

AND

Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode

AND EITHER:

- Event damage has caused indications of degraded performance to the second train of the SAFETY SYSTEM needed for the current operating mode
- Event damage has resulted in VISIBLE DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode

(Notes 9, 10)

Note 9: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then emergency classification is **not** warranted.

Note 10: If the hazardous event **only** resulted in VISIBLE DAMAGE, with **no** indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is **not** warranted.

Table 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 Shift Manager

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

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

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

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

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

VISIBLE DAMAGE - Damage to a SAFETY SYSTEM train that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected SAFETY SYSTEM train.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues.

Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

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

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make a determination of VISIBLE DAMAGE based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

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

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

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

Escalation of the emergency classification level would be via IC FS1 or AS1.

This EAL is the hot condition equivalent of cold condition EAL CA6.1.

Reference(s):

1. EP FAQ 2016-00
2. NEI 99-01 SA9

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.

RBS Table A-3 and H-2 Bases

A review of station operating procedures identified the following mode dependent in-plant actions and associated areas that are required for normal plant operation, cooldown or shutdown:

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
GOP-0005 Power Operations			
Perform power maneuvering as directed by the OSM/CRS between 60 and 100% power using the guidance provided in the appropriate Reactivity Maneuvering Plan provided by Reactor Engineering.	MCR	1	
If possible, notify System Operator prior to changing generator load. • Adhere to MVAR vs. MW limits. • WHEN adjusting VARs on the Main Generator, THEN use VAR-1SPGN05 (H13-P680) only.	MCR	1	
Prior to entry into the Monitored Region of the Power/Flow map verify at least one PBDS Card is operable and begin STP-000-0001 monitoring of PBDS. (TS 3.3.1.3)	MCR	1	
Adjust pressure setpoint to minimize recirc pump "thrust reversals" as follows: • IF lowering power AND it is desired that pressure set be raised to minimize recirc pump "thrust reversals", THEN prior to lowering core flow to less than 70% rated core flow, raise reactor pressure. • IF raising power AND pressure set was raised to minimize recirc pump "thrust reversals", THEN when core flow is greater than 70% rated core flow, return the reactor pressure to its nominal value.	MCR	1	
Monitor Reactor Feed Pump vibration and flow. IF necessary to minimize vibration, THEN operate the reactor feed pump Minimum Flow valves per SOP-0009, Long Cycle Clean Up valve, or adjust reactor power.	MCR	1	
IF a reactor feed pump is anticipated to be shut down and Hydrogen injection will be left in service, THEN install the vent jumper for that pump per SOP-0009, Reactor Feedwater System.	MCR / TB 67'	1	Not required for plant shutdown or cooldown
Remove from service and/or restart Reactor Feed Pumps as necessary to maintain Reactor Water Level and reactor feed pump flow requirements to minimize vibration.	MCR / TB 67'	1	Not required for plant shutdown or cooldown
IF controlling Reactor Power with Reactor Recirculation Flow, THEN refer to SOP-0003.	MCR	1	
IF power is lowered below 75% AND a Reactor Feed Pump has been secured, THEN, BEFORE power ascension beyond 75% RTP and	MCR	1	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
AFTER start of the 3rd feedwater pump verify water chemistry acceptable.			
IF power is lowered below 75%, THEN when thermal power is above 75% RTP, verify both LEFM Check Plus are Operable.	MCR	1	
IF the LEFMs are functional per Technical Requirement 3.3.13 AND an LEFM alert is indicated on the ONE heat balance computer screen, THEN reduce reactor power to 3081 MWth or 99.6% rated thermal power in one hour to ensure thermal power limits are not exceeded.	MCR	1	
Observe the operation MSS-HVYCV4 and DTM-AOVSPDV4 for OSP-0102, Turbine Valve Testing. • As power is raised, check MSS-HVYCV4 open and DTM-AOVSPDV4 closed. • As power is lowered, check MSS-HVYCV4 closed and DTM-AOVSPDV4 open.	MCR	1	
When power is raised above 90%, Pressure Set may need to be adjusted as necessary to ensure that the 1st admission main turbine control valves, MSS-HVYCV1, 2, and 3 are full open.	MCR	1	
Monitor turbine vibration bearing temperature and differential expansion per the following: Turbine Temp & Expansion RCDR (TMI-NXR102) • Differential Expansion Rotor Long (point 11) between 0.31 inches and 0.69 inches. (Refer To ARP-870-54, G08, H08) • Rotor Expansion Rotor Long (point 12) between 0.455 inches and 1.545 inches. Turbine Vibration RCDR (TMI-NXR103) • Vibration (points 1 through 10) between 0 mils and 6 mils. (Refer To ARP-870-54, D08) Tamaris Computer (Display 69, 70) • Bearing oil temperatures (<setpoint, 180°F). • Bearing metal temperatures (<setpoint, 218.7°F). IF unusual indications are observed, THEN initiate hold in power change until those indications return to normal.	MCR	1	
WHEN maneuvering power, THEN adhere to the POWER/FLOW maps (avoid the restricted region) in AOP-0024, Thermal Hydraulic Stability Controls and Turbine-Generator loading rate per SOP-0080, Turbine Generator Operation.	MCR	1	
WHEN in two Recirculation Pump Operation at greater than or equal to 70% rated core flow, THEN maintain recirculation flow mismatch less than 5%.	MCR	1	
Observe the following limitations and precautions: • Do not exceed the Turbine Generator normal operating limits.	MCR	1	

GOP / SOP ACTIONS	LOCATION	MODE	NOTES

• Adjust the pressure setpoint to an indicated reactor pressure of between 1035 and 1055 psig for 100% steady state conditions.
• IF reactor power exceeds 3091 MWth, THEN take actions using recirculation flow and/or control rod insertion to lower power below 3091 MWth.
• IF the core thermal power average for a 2 hour period exceeds the Licensed Power Limit, THEN take timely action to ensure that thermal power is less than or equal to the Licensed Power Limit.
• IF reactor thermal power indication becomes unavailable for less than 15 minutes AND steady state operation is expected, THEN note current APRM readings AND verify thermal power does not exceed the noted value.
• IF reactor thermal power indication will be unavailable for more than 15 minutes, THEN perform the following:
  o Lower reactor power as indicated on the APRMs such that indicated thermal power does not exceed 100%. (The top of the normal noise band on the chart recorders should not be above 100%).
  o Reactor Engineering should be contacted for assistance in determining a manual heat balance per REP-0030, Reactor Heat Balance.
  o WHEN performing a manual heat balance AND it is determined that the LEFM signal is not operable, THEN lower reactor power so that the APRMs read less than 98.3% at the top of the normal noise band.
• Observe the following restrictions when operating near or above rated core flow as Bi-Stable flow conditions are possible:
  o IF step changes of 60 MWth (2%) or greater are seen in instantaneous CTP, THEN reduce Reactor power using Recirc flow until the step changes in instantaneous power are no longer observed.
  o IF step changes of up to 1.69 MLB/hr (2%) are seen in total core flow, THEN reduce Reactor power using Recirc flow until the step changes in instantaneous power are no longer observed.
  o Notify Reactor Engineering of any power/flow reductions required.
• IF any thermal limit exceeds 0.980, THEN notify Reactor Engineering to increase the frequency of monitoring (at least hourly) until a steady state condition is reached or thermal limits indicate less than 0.980.
• IF any thermal limit exceeds 0.990, THEN notify Reactor Engineering to perform one of the following:
  o Provide instructions for reducing the thermal limit to less than 0.990.
  o Provide a justification for operating with thermal limits			

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
greater than or equal to 0.990.			
IF the power change exceeded 15%, THEN perform the following: • Notify Chemistry of the power change to obtain a new Condensate System Oxygen injection flow rate. • Per Chemistry recommendations, adjust the Oxygen flow rate per SOP-0123, Hydrogen Water Chemistry H2 and O2 System. • IF power ramp rates exceed 15%/hr, THEN notify Chemistry per Technical Requirement.	MCR / TB 95'	1	Not required for plant shutdown or cooldown
IF power was lowered below 80%, THEN notify chemistry management when reactor power has been returned to 100%.	MCR	1	
As power is lowered, at approximately 50% power, **transfer Steam Seal Evaporator from Extraction Steam to Main Steam per SOP-0015**, Gland Seal Steam System and Exhaust System, if it has not occurred automatically.	MCR	1	
Transferring Steam Seal Evaporator from Extraction Steam to Main Steam (SOP-0015).			
As power is lowered, at approximately 50% power, if the Steam Seal Evaporator has not already transferred automatically from Extraction Steam to Main Steam, then throttle closed ESS-MOV112, STEAM SEAL EVAPORATOR using the control switch and the STOP pushbutton.	MCR	1	
IF the pressure controller is operating in automatic AND TME-MOVESFV2 is closed, THEN verify the following: • TME-PIEPR-35, SSE TUBE SIDE PRESSURE indicates less than or equal to 75 psig. • TME-PIEPR-36, SSE SHELL SIDE PRESSURE is stable and indicates less than or equal to 45 psig.	MCR	1	
WHEN ESS-MOV112, STEAM SEAL EVAPORATOR is full closed, THEN verify DTM-AOV118, EXTR STM TO SSE & RW RBLR opens.	MCR	1	
As power is raised, at approximately 65-75% power, after checking Annunciator P870-52-E03, 3rd PT EXTR ST AND MAIN STEAM DIFF PRESS LOW is clear, transfer Steam Seal Evaporator from Main Steam to Extraction Steam per SOP-0015, Gland Seal Steam System and Exhaust System.	MCR	1	
GOP-0002 Power Decrease/Plant Shutdown			
Notify System Operator prior to decreasing generator load.	MCR	1	
IF the Reference Leg Backfill System is not in service per SOP-0001, Nuclear Boiler Instrumentation (SYS #051), THEN have I&C stage equipment, acquire necessary technicians and obtain PMs to backfill	MCR	1	

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
reactor water level reference legs. (Approximately 12 hours may be needed to prepare for backfilling.) Actual backfilling performance may commence when Operations Shift Manager authorizes. (It is desired to have backfilling completed prior to reactor pressure reaching 450 psig to counter level indication notching possibilities.)			
Monitor turbine vibration bearing temperature and differential expansion per the following: Turbine Temp & Expansion RCDR (TMI-NXR102) • Differential Expansion Rotor Long (point 11) between 0.31 inches and 0.69 inches. (Refer To ARP-870-54, G08, H08) • Rotor Expansion Rotor Long (point 12) between 0.455 inches and 1.545 inches. Turbine Vibration RCDR (TMI-NXR103) • Vibration (points 1 through 10) between 0 mils and 6 mils. (Refer To ARP-870-54, D08) Tamaris Computer (Display 69, 70) • Bearing oil temperatures (<setpoint, 180°F). • Bearing metal temperatures (<setpoint, 218.7°F). IF unusual indications are observed, THEN initiate hold in power change until those indications return to normal.	MCR	1	
Lower reactor power per the Shutdown/ Emergency Power Reduction reactivity control plan. Contact the on-duty Reactor Engineer.	MCR	1	
Adjust pressure setpoint to minimize recirc pump "thrust reversals" as follows: • IF lowering power AND it is desired that pressure set be raised to minimize recirc pump "thrust reversals", THEN prior to lowering core flow to less than 70% rated core flow, raise reactor pressure. IF raising power AND pressure set was raised to minimize recirc pump "thrust reversals", THEN when core flow is greater than 70% rated core flow, return the reactor pressure to its nominal value.	MCR	1	
At approximately 90% to 80% power observe the following: • MSS-HVYCV4 closes • DTM-AOVSPDV4 opens	MCR	1	
IF MSRs are to be manually shutdown, THEN at approximately 90% power, start removing the MSRs from service per SOP-0010, MSR & FW Heaters Extraction Steam and Drains. Remove the MSRs at a rate so as to be completely off line by 760 MWe. Limit rate of change of LP Turbine inlet steam temperature to 125°F per hour. Monitor Points 6, 7, 8, 9 on TMI-NXR102. Maximum allowable temperature difference between LP Turbine inlets is 50°F. MSRs should be gradually valved	MCR / TB 123'	1	Not required for plant shutdown or cooldown

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
out in parallel at essentially the same temperature. IF MSRs are to remain in service with power maintained between 15% and 70%, THEN operate MSRs in accordance with SOP-0010.			
For power reductions of greater than 15%, notify Chemistry to determine whether the Condensate System oxygen injection is to be secured or flow reduced AND implement the recommendations per SOP-0123, Hydrogen Water Chemistry H2 and O2 System. IF power ramp rates exceed 15%/hr, THEN notify Chemistry per Technical Requirement 3.11.2.1.	MCR / TB 95'	1	Not required for plant shutdown or cooldown
IF a reactor feed pump is anticipated to be shut down AND Hydrogen Injection will be left in service AND a plant shutdown is NOT in progress, THEN install the vent jumper(s) for the pump(s) per SOP-0009, Reactor Feedwater System.	TB 67'	1	Not required for plant shutdown or cooldown
At approximately 70% power, (or with Engineering recommendations) **stop one reactor feedwater pump (leave two running) per SOP-0009, Reactor Feedwater System.**	MCR	1	
Reactor Feed Pump Shutdown (SOP-0009)			
IF securing a Reactor Feed Pump for downpower, THEN monitor the following parameters: • Reactor power should be limited to 85% with only two Reactor Feed Pumps in service. • Normal Feedwater Pump Motor current should be greater than 200 amps and limited to 311 amps. Refer to Precautions and Limitations 2.9 and 2.15. • FWREG position should be limited to less than or equal to 92% open to allow an adequate margin for valve modulation while maintaining reactor level. • Feed pump suction pressure should be maintained above low pressure alarm point of 280 psig.	MCR	1	
IF NOT already performed to reduce Reactor Feed Pump vibration levels, THEN perform the following for the Reactor Feed Pump being shutdown: • At H13-P680, place CNM-H/A68A(B)(C), RX FWP 1A(B)(C) MIN FLOW FLOW CONTROLLER to MANUAL for the Reactor Feed Pump to be secured. • Open slowly FWR-FV2A(B)(C), RX FWP 1A(B)(C) MIN FLOW VALVE using CNM-H/A68A(B)(C), RX FWP 1A(B)(C) MIN FLOW FLOW CONTROLLER while monitoring Reactor Water Level.	MCR	1	
IF desired to raise Reactor Water Level, THEN at H13-P680 adjust C33-R600, FW REG VALVES MASTER FLOW CONTROLLER tape set to desired Reactor Water Level within normal level control band.	MCR	1	

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
IF the HWC System is in service AND the reactor feed pump is not being immediately shut down, THEN at P73-P500, place P73-AOV-F111A(B)(C), HYDROGEN ISOLATION TO FEEDWATER PUMP A(B)(C) in CLOSE.	MCR / TB 95'	1	Not required for plant shutdown or cooldown
IF the capability of meeting feed flow requirements with the remaining Feedwater Pumps is uncertain, THEN make a determination as follows: • Close FWS-MOV26A(B)(C), RX FWP 1A(B)(C) DISCH VLV for the pump being shutdown. • Verify the minimum flow valve for the pump being secured is open. • Monitor Feed Flow/Steam Flow mismatch and RPV Level to ensure remaining pump(s) can maintain level. • IF the remaining pump(s) cannot maintain RPV Level, THEN reopen the discharge valve FWS-MOV26A(B)(C), RX FWP 1A(B)(C) DISCH VLV and discontinue this procedure.	MCR	1	
IF the last Feedwater Pump is being removed from service, THEN open FWS-MOV109, FEED PUMP BYPASS.	MCR	1	
Stop FWS-P1A(B)(C), RX FWP P1A(B)(C).	MCR	1	
Verify CNM-H/A68A(B)(C), RX FWP 1A(B)(C) MIN FLOW FLOW CONTROLLER is in AUTO for the Reactor Feed Pump that was secured.	MCR	1	
IF Reactor Water Level was intentionally raised in Step 6.1.3, THEN adjust Reactor Water Level to desired level within normal level control band using C33-R600, FW REG VALVES MASTER FLOW CONTROLLER tape set.	MCR	1	

IF FWS-P1A(B)(C) is to remain in hot standby, THEN maintain seal temperatures as follows:
Maintain seal water temperature dT less than or equal to 50F AND seal water outlet temperature less than or equal to 300F as follows:
• FWS-P1A
  o Throttle CCS-V5003A, RFP FWL-P1A SEAL WATER HX-E4A CCS INLET VALVE, as required.
  o Throttle CCS-V5004A, RFP FWL-P1A SEAL WATER HX-E4B CCS INLET VALVE, as required.
• FWS-P1B
  o Throttle CCS-V5003B, RFP FWL-P1B SEAL WATER HX-E4C CCS INLET VALVE, as required.
  o Throttle CCS-V5004B, RFP FWL-P1B SEAL WATER HX-E4D CCS INLET VALVE, as required.
• FWS-P1C
  o Throttle CCS-V5003C, RFP FWL-P1C SEAL WATER HX-E4E CCS INLET VALVE, as required.
  o Throttle CCS-V5004C, RFP FWL-P1C SEAL WATER
	TB 67'	1	Not required for plant shutdown or cooldown

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
HX-E4F CCS INLET VALVE, as required.			
On H13-P870, verify FWL-P5A(B)(C), GEAR INCR AUX OIL PMP 5A(B)(C) auto starts.	MCR	1	
Verify min flow valve closes 1 - 3 minutes after pump shutdown.	MCR	1	
Verify FWS-MOV26A(B)(C), RX FWP P1A (B)(C) DISCH VLV is closed.	MCR	1	
On H13-P870, WHEN the 23 minute time delay allowing for pump coast down has passed, THEN verify the following: • FWL-P1A(B)(C), RX FWP A(B)(C) MN OIL PMP 1A(B)(C) auto stops. • FWL-P5A(B)(C), RX FWP A(B)(C) GEAR INC AUX OIL PMP 5A(B)(C) auto stops. • FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) auto starts on low oil pressure. • IF FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) does not maintain pressure greater than 4 psi, THEN FWL-P3A(B)(C), RX FWP A(B)(C) AUX OIL PMP 3A(B)(C) auto starts on low oil pressure.	MCR	1	
On H13-P870, place FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) control switch in STOP, and verify FWL-P3A(B)(C), RX FWP A(B)(C) AUX OIL PMP 3A(B)(C) auto starts on low oil pressure.	MCR	1	
On H13-P870, place FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) control switch in AUTO.	MCR	1	
On H13-P870, place FWL-P3A(B)(C), RX FWP A(B)(C) AUX OIL PMP 3A(B)(C) control switch in STOP.	MCR	1	
On H13-P870, place FWL-P3A(B)(C), RX FWP A(B)(C) AUX OIL PMP 3A(B)(C) control switch in AUTO.	MCR	1	
Locally verify breaker relay trip flags are reset for Reactor Feed Pump stopped in Step 6.1.7.	NSW 98'	1	Not required for plant shutdown or cooldown
At approximately 70% power, (or with Engineering recommendations) **stop one condensate pump (leave two running) per SOP-0007, Condensate System.**	MCR	1	
Shutdown of CNM-P1A(B)(C) CONDENSATE PUMPS (SOP-0007)			
Request Aux Control Room remove unnecessary Condensate Filters from service per SOP-0124, Condensate Filtration System.	ACR	1	Not required for plant shutdown or cooldown
Request Aux Control Room remove unnecessary Condensate Demins from service per SOP-0093, Condensate Demineralizer System.	ACR	1	Not required for plant shutdown or

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
			cooldown
IF shutting down CNM-P1C, CNDS PUMP 1C, THEN secure Oxygen injection per SOP-0123, Hydrogen Water Chemistry H2 and O2 System.	MCR / TB 95'	1	Not required for plant shutdown or cooldown
IF securing the Condensate System, THEN perform the following: • Close all CNM-V3105 A, B, C, D, and E, CNM-FLT1 A, B, C, D, and E BACKWASH AIR SUPPLY valves. • IF desired to isolate and depressurize CNM-TK100, AIR RECEIVING TK, THEN perform the following: o CLOSE CNM-V3110, SVCE AIR ISOL VLV INLET SERV. AIR ISOL VLV. o Uncap and install hose on CNM-V3112, CNM-TK100 DRAIN ISOLATION VALVE. o Open CNM-V3112.	TB 123'	1	Not required for plant shutdown or cooldown
Depress the CLOSE pushbutton for CNM-MOV3A(B)(C), CNDS PUMP 1A(1B)(1C) DISCH.	MCR	1	
WHEN pump motor current lowers below 100 amps, THEN stop CNM-P1A(B)(C), CNDS PUMP 1A(1B)(1C).	MCR	1	
WHEN CNM-MOV3A(B)(C), CNDS PUMP 1A(1B)(1C) DISCH is full closed, THEN depress the STOP pushbutton.	MCR	1	
Verify associated CCS-MOV67A(B)(C), CNDS PMP 1A(1B)(1C) MOT CLR close for pump stopped.	MCR	1	
Verify associated CCS-MOV68A(B)(C), CNDS PMP 1A(1B)(1C) BRG CLR close for pump stopped.	MCR	1	
Locally verify breaker relay trip flags are reset for Condensate Pump stopped in Step 6.1.6.	NSW 98'	1	Not required for plant shutdown or cooldown
WHEN the Steam Jet Air Ejectors (SJAEs) and Gland Seal and Exhaust System are removed from service, THEN adjust CNM-H/A114 to 10% or to a setpoint determined by the CRS/OSM.	MCR	1	
As power is reduced, **remove FW Reg Valves from service per SOP-0009,** Reactor Feedwater System.	MCR	1	
Removing a FWREG Valve from Service (SOP-0009)			
Check feedwater flow is within the capability of the remaining FWREGs.	MCR	1	
Station an operator locally at the FWREG Valve to be removed from service.	TB 67'	1	Not required for plant shutdown or cooldown

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Establish communications between the local operator and the Main Control Room (MCR).	MCR / TB 67'	1	Not required for plant shutdown or cooldown
Place C33-R601A(R613)(R601B), FWREG VALVE A(B)(C) FLOW CONTROLLER in MANUAL.	MCR	1	
Throttle closed to 10% open the C33-LVF001A(B)(C), FWREG VALVE A(B)(C) to be removed from service while observing that reactor level is being maintained by the remaining FWREGs.	MCR	1	
IF level is not being maintained by the remaining FWREGs, THEN place the FWREG that was being removed from service back in service as follows: • Open C33-LVF001A(B)(C), FWREG VALVE A(B)(C) to the same position as the in service FWREGs. • Place C33-R601A(R613)(R601B), FWREG VALVE A(B)(C) FLOW CONTROLLER in AUTO.	MCR	1	
WHEN the FWREG is at 10% open, THEN close the following isolation valve for the FWREG valve that is being removed from service. • For C33-LVF001A close FWS-MOV27A, FWREG VLV 1A INLT Valve. • For C33-LVF001B close FWS-MOV27B, FWREG VLV 1B INLT Valve. • For C33-LVF001C close FWS-MOV27C, FWREG VLV 1C INLT Valve.	MCR	1	
Fully close the C33-LVF001A(B)(C), FWREG VALVE A(B)(C) that was removed from service.	MCR	1	
Record the temperature of the feedwater at the reactor feed pumps.	TB 67'	1	Not required for plant shutdown or cooldown
IF FWS-MOV27A, B, or C, FWREG VLV 1A(1B)(1C) INLT were closed with feedwater temperature at the reactor feed pumps greater than 200F, THEN refer to Section 5.7 for further stroking requirements.	MCR	1	
WHEN the FWREG Valve is at 0% open, THEN record demanded position in the MCR, position indication in the MCR, and local position indication.	MCR / TB 67'	1	Not required for plant shutdown or cooldown
Within one hour after reactor power is less than or equal to the high power setpoint, demonstrate RWL operability by performing STP-500-0704, Rod Withdrawal Limiter Functional Test (SR 3.3.2.1.2), if not performed within the previous 92 days.	MCR	1	STP-500-0704, Rod Withdrawal Limiter Functional Test is

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
			performed only in the MCR
Prior to entering the Monitored and/or the Restricted Regions of the Power to Flow map, verify the following indications on the PBDS, Period Based Detection System cards in APRM 'A' and 'B' cabinets: • NORMAL/BYPASS Toggle switch in the NORMAL position. • INOP STATUS LED indication is GREEN. (Depress the INOP STATUS Reset Pushbutton to reset a Red LED inop indication.) • Verify at least one PBDS Card is OPERABLE. • Begin STP-000-0001 monitoring of PBDS.	MCR	1	
Prior to entry into the Restricted Region of the Power to Flow Map, perform the following: • Verify FCBB is less than or equal to 1.0.(SR 3.2.4.1) • Place the APRM - FCTR, Flow Control Trip Reference cards to the setup trip setpoints by depressing the Normal/Setup pushbutton and verifying the normal/setup LED indication is yellow.	MCR	1	
At Approximately 50% power, **transfer Steam Seal Evaporator from Extraction Steam to Main Steam per SOP-0015, Gland Seal System And Exhaust System,** if it has not occurred automatically.	MCR	1	
Transferring Steam Seal Evaporator from Extraction Steam to Main Steam (SOP-0015)			
As power is lowered, at approximately 50% power, if the Steam Seal Evaporator has not already transferred automatically from Extraction Steam to Main Steam, then throttle closed ESS-MOV112, STEAM SEAL EVAPORATOR using the control switch and the STOP pushbutton.	MCR	1	
IF the pressure controller is operating in automatic AND TME-MOVESFV2 is closed, THEN verify the following: • TME-PIEPR-35, SSE TUBE SIDE PRESSURE indicates less than or equal to 75 psig. • TME-PIEPR-36, SSE SHELL SIDE PRESSURE is stable and indicates less than or equal to 45 psig.	MCR	1	
WHEN ESS-MOV112, STEAM SEAL EVAPORATOR is full closed, THEN verify DTM-AOV118, EXTR STM TO SSE & RW RBLR opens.	MCR	1	
At approximately 50% power, **shutdown all heater drain pumps per SOP-0010, MSR & FW Heaters Extraction Steam and Drains**.	MCR	1	
Removing the Heater Drain Pumps From Service (SOP-0010)			

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
IF it is desired to secure HDL-P1A(B), HTR DR PUMP 1A(B) for Heater String A, THEN at H13-P680, perform the following: • Depress the Close Pushbutton for HDL-MOV55A(B), HTR DR PMP 1A(B) DISCH. • Stop HDL-P1A(B), HTR DR PUMP 1A(B) for Heater String A. • Verify HDL-MOV55A(B), HTR DR PMP 1A(B) DISCH is closed.	MCR	1	
IF it is desired to secure HDL-P1C(D), HTR DR PUMP 1C(D) for Heater String B, THEN at H13-P680, perform the following: • Depress the Close Pushbutton HDL-MOV55C(D), HTR DR PMP 1C(D) DISCH. • Stop HDL-P1C(D), HTR DR PUMP 1C(D) for Heater String B. • Verify HDL-MOV55C(D), HTR DR PMP 1C(D) DISCH is closed.	MCR	1	
At approximately 50% power, perform the following per SOP-0006, Circulating Water, Cooling Tower and Vacuum Priming: • Shut down at least 1 circulating water pump. • Adjust the number of operating cooling tower fans to maintain vacuum and circulating water temperature.	MCR	1	
WHEN the recirculation flow control valves are at their minimum position, THEN continue reducing power by inserting control rods in their proper sequence.	MCR	1	
At about 40% power, **transfer both reactor recirculation pumps to SLOW speed per SOP-0003, Reactor Recirculation System.**	MCR	1	
Transferring from Fast Speed to Slow Speed (SOP-0003)			
Simultaneously depress B33-C001A and B RECIRC PUMP A and B MOTOR BREAKER 5A and 5B XFER TO LFMG pushbuttons.	MCR	1	
Observe the following: • Both B33-S001A LFMG MOT BRKR 1A and B33-S001B LFMG MOTBRKR 1B close. • Both B33-C001A RECIRC PUMP A MOTOR BREAKER 5A and B33-C001B RECIRC PUMP B MOTOR BREAKER 5B open. • WHEN B33-C001A and B, RECIRC PUMP A and B coast down to approximately 360 - 470 RPM, THEN B33-S001A and B LFMG A and B GEN BRKR 2A and 2B close and pump speeds stabilize near 450 RPM. • Both B33-K603 A and B, RECIRC LOOP A and B FLOW CONTROL MAN/AUTO stations transfer to MAN. • Open B33-HVY-F060A(B) to approximately 94% valve position using B33-K603A(B).	MCR	1	
Reduce to one reactor feed pump per SOP-0009, Reactor Feedwater System.	MCR	1	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Reactor Feed Pump Shutdown (SOP-0009)			
IF securing a Reactor Feed Pump for downpower, THEN monitor the following parameters: • Reactor power should be limited to 85% with only two Reactor Feed Pumps in service. • Normal Feedwater Pump Motor current should be greater than 200 amps and limited to 311 amps. Refer To Precautions and Limitations 2.9 and 2.15. • FWREG position should be limited to less than or equal to 92% open to allow an adequate margin for valve modulation while maintaining reactor level. • Feed pump suction pressure should be maintained above low pressure alarm point of 280 psig.	MCR	1	
IF NOT already performed to reduce Reactor Feed Pump vibration levels, THEN perform the following for the Reactor Feed Pump being shutdown: • At H13-P680, place CNM-H/A68A(B)(C), RX FWP 1A(B)(C) MIN FLOW FLOW CONTROLLER to MANUAL for the Reactor Feed Pump to be secured. • Open slowly FWR-FV2A(B)(C), RX FWP 1A(B)(C) MIN FLOW VALVE using CNM-H/A68A(B)(C), RX FWP 1A(B)(C) MIN FLOW FLOW CONTROLLER while monitoring Reactor Water Level.	MCR	1	
IF desired to raise Reactor Water Level, THEN at H13-P680 adjust C33-R600, FW REG VALVES MASTER FLOW CONTROLLER tape set to desired Reactor Water Level within normal level control band.	MCR	1	
IF the HWC System is in service AND the reactor feed pump is not being immediately shut down, THEN at P73-P500, place P73-AOV-F111A(B)(C), HYDROGEN ISOLATION TO FEEDWATER PUMP A(B)(C) in CLOSE.	MCR / TB 95'	1	Not required for plant shutdown or cooldown
IF the capability of meeting feed flow requirements with the remaining Feedwater Pumps is uncertain, THEN make a determination as follows: • Close FWS-MOV26A(B)(C), RX FWP 1A(B)(C) DISCH VLV for the pump being shutdown. • Verify the minimum flow valve for the pump being secured is open. • Monitor Feed Flow/Steam Flow mismatch and RPV Level to ensure remaining pump(s) can maintain level. • IF the remaining pump(s) cannot maintain RPV Level, THEN reopen the discharge valve FWS-MOV26A(B)(C), RX FWP 1A(B)(C) DISCH VLV and discontinue this procedure.	MCR	1	
IF the last Feedwater Pump is being removed from service, THEN open FWS-MOV109, FEED PUMP BYPASS.	MCR	1	
Stop FWS-P1A(B)(C), RX FWP P1A(B)(C).	MCR	1	

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Verify CNM-H/A68A(B)(C), RX FWP 1A(B)(C) MIN FLOW FLOW CONTROLLER is in AUTO for the Reactor Feed Pump that was secured.	MCR	3	
IF Reactor Water Level was intentionally raised in Step 6.1.3, THEN adjust Reactor Water Level to desired level within normal level control band using C33-R600, FW REG VALVES MASTER FLOW CONTROLLER tape set.	MCR	3	
IF FWS-P1A(B)(C) is to remain in hot standby, THEN maintain seal temperatures as follows: Maintain seal water temperature dT less than or equal to 50F AND seal water outlet temperature less than or equal to 300F as follows: • FWS-P1A o Throttle CCS-V5003A, RFP FWL-P1A SEAL WATER HX-E4A CCS INLET VALVE, as required. o Throttle CCS-V5004A, RFP FWL-P1A SEAL WATER HX-E4B CCS INLET VALVE, as required. • FWS-P1B o Throttle CCS-V5003B, RFP FWL-P1B SEAL WATER HX-E4C CCS INLET VALVE, as required. o Throttle CCS-V5004B, RFP FWL-P1B SEAL WATER HX-E4D CCS INLET VALVE, as required. • FWS-P1C o Throttle CCS-V5003C, RFP FWL-P1C SEAL WATER HX-E4E CCS INLET VALVE, as required. o Throttle CCS-V5004C, RFP FWL-P1C SEAL WATER HX-E4F CCS INLET VALVE, as required.	TB 67'	1	Not required for plant shutdown or cooldown
On H13-P870, verify FWL-P5A(B)(C), GEAR INCR AUX OIL PMP 5A(B)(C) auto starts.	MCR	1	
Verify min flow valve closes 1 - 3 minutes after pump shutdown.	MCR	1	
Verify FWS-MOV26A(B)(C), RX FWP P1A (B)(C) DISCH VLV is closed.	MCR	1	
On H13-P870, WHEN the 23 minute time delay allowing for pump coast down has passed, THEN verify the following: • FWL-P1A(B)(C), RX FWP A(B)(C) MN OIL PMP 1A(B)(C) auto stops. • FWL-P5A(B)(C), RX FWP A(B)(C) GEAR INC AUX OIL PMP 5A(B)(C) auto stops. • FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) auto starts on low oil pressure. • IF FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) does not maintain pressure greater than 4 psi, THEN FWL-P3A(B)(C), RX FWP A(B)(C) AUX OIL PMP 3A(B)(C) auto starts on low oil pressure.	MCR	1	
On H13-P870, place FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) control switch in STOP, and verify FWL-P3A(B)(C), RX FWP	MCR	1	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
A(B)(C) AUX OIL PMP 3A(B)(C) auto starts on low oil pressure.			
On H13-P870, place FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) control switch in AUTO.	MCR	1	
On H13-P870, place FWL-P3A(B)(C), RX FWP A(B)(C) AUX OIL PMP 3A(B)(C) control switch in STOP.	MCR	1	
On H13-P870, place FWL-P3A(B)(C), RX FWP A(B)(C) AUX OIL PMP 3A(B)(C) control switch in AUTO.	MCR	1	
Locally verify breaker relay trip flags are reset for Reactor Feed Pump stopped in Step 6.1.7.	NSW 98'	1	Not required for plant shutdown or cooldown
Decrease reactor power at the rate consistent with generator loading criteria (Attachment 1, MVAR VS MW LIMITS and SOP-0080, Turbine Generator Operation) using control rod insertion per applicable sequence. • Stop inserting control rods at the low power alarm point (as observed in RC & IS panel) and obtain instruction from the Operations Shift Manager regarding further power reductions/shutdown or continued operation at LPAP. • Decrease reactor power to the LPSP using control rod insertion per applicable sequence.	MCR	1	
At 300 MWe load, open the following steam drain valves: • DTM-AOV32A, 4TH PT HTR EXTR LINE DR • DTM-AOV32B, 4TH PT HTR EXTR LINE DR • DTM-AOV35A, 3RD PT HTR EXTR LINE DR • DTM-AOV35B, 3RD PT HTR EXTR LINE DR	MCR	1	
Open or verify open G33-MOVF101, RWCU BOTTOM HEAD DRAIN. • Verify drain temperature remains stable using Point #4 on B21R643 or ERIS computer point B33NA002.	MCR	1	
Manually stroke C33-LVF002, STARTUP FWREG VALVE through full travel to verify smooth operation **per SOP-0009, Reactor Feedwater System.**	MCR	1	
Manual Stroking of Start Up FWREG (SOP-0009)			
Close FWS-MOV105, S/U FW REG VLV ISOL.	MCR	1	
Station an operator locally to monitor valve position.	TB 67'	1	Not required for plant shutdown or cooldown

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Establish communications between the local operator and the Main Control Room (MCR).	MCR / TB 67'	1	Not required for plant shutdown or cooldown
WHEN the FWREG Valve is at 0% open, THEN record demanded position in the MCR, position indication in the MCR, and local position indication in Attachment 7, Calibration Check of FWREG Valves.	MCR / TB 67'	1	Not required for plant shutdown or cooldown
Use the OPEN and CLOSE Pushbuttons on C33-R602, START UP FWREG VALVE FLOW CONTROLLER to stroke open and then closed the Start Up FWREG.	MCR	1	
Check proper valve movement and smooth operation.	MCR / TB 67'	1	Not required for plant shutdown or cooldown
Check C33-LVF002, START UP FWREG VLV full closed.	MCR	1	
Open FWS-MOV105, S/U FW REG VLV ISOL.	MCR	1	
WHEN less than 30% power AND at the direction of the responsible Operations Management, THEN perform the following: • Transfer station loads to preferred source per SOP-0045, 13.8 KV System and SOP-0046, 4.16 KV System. • Verify MVARs are between + 50 and – 50. • At the SRM cabinets, place the Mode Selector Switches to the OPERATE position. • Prior to initiating a Rx Scram, verify the SRM & IRM Channel Functional Tests are current. IF Channel Functional Tests are not current, THEN refer to Tech Spec 3.3.1.1, 3.3.1.2 and TRM TR 3.3.2.1.	MCR	1	
Secure SPC per SOP-0140, Suppression Pool Cleanup and Alternate Decay Heat Removal.	MCR	1	Not required, but system will automaticall y isolate on a level 3 from a RX SCRAM
Contact the Auxiliary Control Room to verify that sufficient condensate demineralizers are in service to prevent physical damage to the demineralizers from high feedwater flow transients.	MCR / ACR	1/3	Not required for plant shutdown or cooldown

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Contact the Auxiliary Control Room to verify that sufficient condensate filtration filters are in service to prevent physical damage to the filters from high feedwater flow transients	MCR / ACR	1/3	Not required for plant shutdown or cooldown
Reduce the number of FWREG Valves in service to one per SOP-0009, Reactor Feedwater System.	MCR	1	Not required for plant shutdown or cooldown
Removing a FWREG Valve from Service (SOP-0009)			
Check feedwater flow is within the capability of the remaining FWREGs.	MCR	1	
Station an operator locally at the FWREG Valve to be removed from service.	TB 67'	1	Not required for plant shutdown or cooldown
Establish communications between the local operator and the Main Control Room (MCR).	MCR / TB 67'	1	Not required for plant shutdown or cooldown
Place C33-R601A(R613)(R601B), FWREG VALVE A(B)(C) FLOW CONTROLLER in MANUAL.	MCR	1	
Throttle closed to 10% open the C33-LVF001A(B)(C), FWREG VALVE A(B)(C) to be removed from service while observing that reactor level is being maintained by the remaining FWREGs.	MCR	1	
IF level is not being maintained by the remaining FWREGs, THEN place the FWREG that was being removed from service back in service as follows: • Open C33- LVF001A(B)(C), FWREG VALVE A(B)(C) to the same position as the in service FWREGs. • Place C33-R601A(R613)(R601B), FWREG VALVE A(B)(C) FLOW CONTROLLER in AUTO.	MCR	1	
WHEN the FWREG is at 10% open, THEN close the following isolation valve for the FWREG valve that is being removed from service • For C33-LVF001A close FWS-MOV27A, FWREG VLV 1A INLT Valve. • For C33-LVF001B close FWS-MOV27B, FWREG VLV 1B INLT Valve. • For C33-LVF001C close FWS-MOV27C, FWREG VLV 1C INLT Valve.	MCR	1	
Fully close the C33-LVF001A(B)(C), FWREG VALVE A(B)(C) that was removed from service.	MCR	1	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Record the temperature of the feedwater at the reactor feed pumps.	TB 67'	1	
IF FWS-MOV27A, B, or C, FWREG VLV 1A(1B)(1C) INLT were closed with feedwater temperature at the reactor feed pumps greater than 200F, THEN refer to Section 5.7 for further stroking requirements.	MCR	1	
WHEN the FWREG Valve is at 0% open, THEN record demanded position in the MCR, position indication in the MCR, and local position indication.	MCR / TB 67'	1	Not required for plant shutdown or cooldown
Line up RWCU reject to the main condenser **per SOP-0090, Reactor Feedwater System.**	MCR	1	Not required, but the preferred method to control level if shutdown long term.
RWCU blowdown operations (SOP-0090)			
Request chemistry sample to verify reactor water quality is within the specifications of Technical Requirement 3.4.13.	MCR	1	
Notify Radiation Protection prior to rejecting water to the Main Condenser or Radwaste.	MCR	1	
IF rejecting to the Main Condenser, THEN open G33 F046, RWCU DRAIN TO MN COND.	MCR	1	
IF rejecting during cold shutdown or refueling, THEN open G33 F031, RWCU DRAIN ORIFICE BYP.	MCR	1	
IF rejecting with the RWCU HXs isolated, THEN perform the following: • Open G33-F107, RWCU REGEN HX BYPASS. • Throttle open G33-PVF033, RWCU REJECT FLOW VALVE to establish reject flow as indicated on G33-R602, RWCU REJECT FLOW. • IF necessary to establish adequate reject flow, THEN close G33-F040, RWCU INBD RETURN VALVE.	MCR	1	
To establish the reject and maintain RWCU flowrate on G33 R609, RWCU INLET FLOW nearly constant, simultaneously throttle the following: • G33-PVF033, RWCU REJECT FLOW VALVE open using G33 R606, RWCU REJECT FLOW CONTROLLER • G33 F042, RWCU REGEN HX OUTLET closed	MCR	1	
Observe blowdown flow on G33 R602, RWCU REJECT FLOW.	MCR	1	
Monitor reactor water level while blowdown is in progress.	MCR	1	

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
IF desired, THEN shutdown Reactor Recirculation HPU A(B) per SOP-0003 to prevent unnecessary Flow Control Valve movement.	MCR	1	
Initiate a Manual Scram per AOP-0001, Reactor Scram. • Verify the Hydrogen Water Chemistry (HWC) System shuts down on scram signal.	MCR	1/3	
WHEN it is desired to bypass the Feedwater Pump Level 8 Trip, THEN perform Attachment 5, Feedwater Pump Level 8 Trip Jumper Installation/Restoration Step 1.	MCR	3	Not required for plant shutdown or cooldown
WHEN it is desired to bypass the MSO Level 8 Trip, THEN perform Attachment 8, MSO Level 8 BYPASS Switch Step 1.	MCR	3	Not required for plant shutdown or cooldown
Monitor Bottom Head Drain Temperature on B21-R643 Point 4 or B33NA002 and take the following actions, as necessary, in a timely controlled manner to prevent an excessive temperature change. (STP-050-0700, RCS Pressure/Temperature Limits Verification). • Reset the Scram. • Reset any FCV runback per ARP-680-04.	MCR	3	
Within one hour after THERMAL POWER < 10% RTP in MODE 1, complete the following steps: • Verify/ensure that the RCIS data mode is selected to "CHAN 1 and CHAN 2". • Select and attempt to withdraw an out-of-sequence control rod. • Verify no rod motion occurs. • Verify Annunciator, P680-07A-C01, CONTROL ROD WITHDRAWAL BLOCK is actuated. • Verify WITHDRAWAL BLOCK Status Light is ON and not flashing. (SR 3.3.2.1.4)	MCR	3	
Place the APRM FCTR Cards to the Normal trip setpoints by depressing the Normal/Setup pushbutton and verifying the normal/setup LED indication is green	MCR	3	
After the Main Turbine is tripped, open the Feedwater Heater Vents per SOP-0010, MSR & FW Heaters Extraction Steam and Drains.	MCR	3	
Establish MSR Steam Blanketing per SOP-0010, MSR & FW Heaters Extraction Steam and Drains.	MCR	3	
Establishing Steam Blanketing (SOP-0010)			
IF Aux. Steam is available, THEN perform the following: • Throttle ASR-MOV104, MSR STM BLANKET SHUTOFF	MCR	3	

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
open until both red and green indication is received. • WHEN Aux steam to MSR steam blanketing line is warm as indicated on Computer point ASRTA01, THEN fully open ASR-MOV104. Verify the following are closed: • MSS-MOV111, MSR 1 STM SPLY SHUTOFF • MSS-MOV112, MSR 2 STM SPLY SHUTOFF • MSS-PVRSHLV1, MSR 1 HIGH LOAD VALVE • MSS-PVRSHLV2, MSR 2 HIGH LOAD VALVE • MSS-PVRSLLV1, MSR 1 LOW LOAD VALVE • MSS-PVRSLLV2, MSR 2 LOW LOAD VALVE • DSR-MOV107, SCAV STM TO 1ST PT HTR A • DSR-MOV109, SCAV STM TO 1ST PT HTR B • DSR-MOV108, SCAV STM TO COND A • 10) DSR-MOV110, SCAV STM TO COND B • DTM-MOV54A, MSL TO MSR 1 COND DR • DTM-MOV54B, MSL TO MSR 2 COND DR IF Aux Steam is available, THEN open the following: • ASR-MOVBSFV1, MSR 1 STM BLANKET SPLY • ASR-MOVBSFV2, MSR 2 STM BLANKET SPLY WHEN several minutes have elapsed after opening ASR-MOVBSFV1 and ASR-MOVBSFV2, THEN place the following control switches to CLOSE: • MSS-MOV111, MSR 1 STM SPLY SHUTOFF • MSS-MOV112, MSR 2 STM SPLY SHUTOFF			
IF notching is observed during the depressurization and magnitude is less than six inches, THEN: • Make all possible attempts to maintain reactor pressure. • Have I&C backfill the reference leg in which notching was observed, even if reference leg was overfilled prior to this event. IF notching is observed during the depressurization and magnitude is greater than six inches, THEN declare the trip channels associated with that signal inoperable and comply with Technical Specification requirements.	MCR	3	Not required for plant shutdown or cooldown
Reduce the number of running condensate pumps to one per SOP-0007, Condensate System.	MCR	3	
Shutdown of CNM-P1A(B)(C) CONDENSATE PUMPS (SOP-0007)			
Request Aux Control Room remove unnecessary Condensate Filters from service per SOP-0124, Condensate Filtration System.	ACR	1	Not required for plant shutdown or cooldown

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Request Aux Control Room remove unnecessary Condensate Demins from service per SOP-0093, Condensate Demineralizer System.	ACR	1	Not required for plant shutdown or cooldown
IF shutting down CNM-P1C, CNDS PUMP 1C, THEN secure Oxygen injection per SOP-0123, Hydrogen Water Chemistry H2 and O2 System.	MCR / TB 95'	1	Not required for plant shutdown or cooldown
IF securing the Condensate System, THEN perform the following: • Close all CNM-V3105 A, B, C, D, and E, CNM-FLT1 A, B, C, D, and E BACKWASH AIR SUPPLY valves. • IF desired to isolate and depressurize CNM-TK100, AIR RECEIVING TK, THEN perform the following: o CLOSE CNM-V3110, SVCE AIR ISOL VLV INLET SERV. AIR ISOL VLV. o Uncap and install hose on CNM-V3112, CNM-TK100 DRAIN ISOLATION VALVE. o Open CNM-V3112.	TB 123'	1	Not required for plant shutdown or cooldown
Depress the CLOSE pushbutton for CNM-MOV3A(B)(C), CNDS PUMP 1A(1B)(1C) DISCH.	MCR	1	
WHEN pump motor current lowers below 100 amps, THEN stop CNM-P1A(B)(C), CNDS PUMP 1A(1B)(1C).	MCR	1	
WHEN CNM-MOV3A(B)(C), CNDS PUMP 1A(1B)(1C) DISCH is full closed, THEN depress the STOP pushbutton.	MCR	1	
Verify associated CCS-MOV67A(B)(C), CNDS PMP 1A(1B)(1C) MOT CLR close for pump stopped.	MCR	1	
Verify associated CCS-MOV68A(B)(C), CNDS PMP 1A(1B)(1C) BRG CLR close for pump stopped.	MCR	1	
Locally verify breaker relay trip flags are reset for Condensate Pump stopped in Step 6.1.6	NSW 98'		Not required for plant shutdown or cooldown
WHEN the Steam Jet Air Ejectors (SJAEs) and Gland Seal and Exhaust System are removed from service, THEN adjust CNM-H/A114 to 10% or to a setpoint determined by the CRS/OSM.	MCR	1	
WHEN less than or equal to 145 MWT, THEN perform the following: • Secure SJAE per SOP-0092, Offgas System.	MCR	1	
• Start a mechanical vacuum pump per SOP-0025, Condenser Air Removal System.	TB 123' & 95'		Not required for plant shutdown or

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
			cooldown
WHEN all rods have been fully inserted, THEN initiate RPV cooldown at less than or equal to 100°F/hr by one of the following methods: • Initiate an automatic cooldown of the RPV from HMI Screen 5532, Pressure Control by performing the following: o Depress AUTO RATE in the Cooldown/Heatup Rate section and verify the pushbutton turns cornsilk and disabled. o Entered the desired cooldown rate and depress ENTER. o Enter the desired Target Pressure in the Throttle Pressure Control section and verify the Target pressure reflects the value entered. o Depress the GO pushbutton and verify: ▪ Pressure regulator setpoint is changing automatically. ▪ Bypass valves modulate to control pressure. ▪ RPV temperature lowers in accordance with the selected cooldown rate. • Initiate a cooldown by slowing and periodically reducing turbine pressure regulator setpoint from HMI Screen 5532, Pressure Control as follows: o Enter the desired Target pressure into the HMI display and depress ENTER. o Depress the GO pushbutton and verify Press Set changes to reflect the Target value. • ON HMI Screed 5537, BPV Jack, control the Bypass valves as follows to establish the required steam flow to the main condenser for cooldown: o Enter the BPV Jack Rate and depress ENTER. o Enter the BPV Target and depress ENTER. o Verify the BPVs open to the desired Target value to establish steam flow to the main condenser.	MCR	3	
Prior to reaching 600 psig, initiate monitoring of ERIS Narrow Range level for "notching" of one or more level indications (IF ERIS trending is not available, THEN contact I&C to arrange for alternate Narrow Range trending).	MCR	3	
Down range IRMs to maintain indication between downscale alarm and upscale alarm.	MCR	3	
Insert SRMs to maintain SRM counts between 1 x 103 and 1 x 105 cps. Fully insert SRMs before the IRMs are on range 3.	MCR	3	
Verify overlap between SRM and IRM. (All SRMs reading < 1x105 cps	MCR	3	

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
prior to IRMs reading < 5 on range 1.)			

WHEN all control rods are fully inserted, THEN perform one of the following:
• NORMAL SHUTDOWN
  o Place the REACTOR SYSTEM MODE SWITCH to SHUTDOWN.
  o WHEN at least 10 seconds have elapsed, THEN reset the Reactor Scram.
• SOFT SHUTDOWN
  o Bypass the REACTOR MODE SWITCH POSITION SCRAM per SOP-0079, Reactor Protection System Attachment 7.
  o Place the REACTOR SYSTEM MODE SWITCH to SHUTDOWN.
  o WHEN at least 10 seconds have elapsed, THEN restore the REACTOR MODE SWITCH POSITION SCRAM per SOP-0079, Reactor Protection System Attachment 7.
	MCR	3	

WHEN the reactor is shutdown AND at the direction of the Operations Shift Manager, THEN perform a drywell inspection per Attachment 3, Drywell Inspection Checklist.	DW 141'/131'/118'/107'/95'/82'	3	Not required for plant shutdown or cooldown
Maintain hot shutdown condition with RPV pressure between 250 psig and 1055 psig.	MCR	3	
IF RHR Pump warmup and flushing is required, THEN perform warmup/flushing per SOP-0031, Residual Heat Removal.	MCR	3	
Shutdown Cooling Flush, Warmup, and Startup (SOP-0031)			
IF RPV pressure is less than 135 psig, THEN have Electrical Maintenance implement the PM Task to re-land the thermal overload/loss of power annunciator leads for E12-F009, RHR SHUTDOWN COOLING INBD ISOL VALVE.	MCR	3	
Verify the following breakers are ON: • EHS-MCC2E BKR 5C, C002A DISCH MIN FLOW VALVE • EHS-MCC2F BKR 7B, C002B DISCH MIN FLOW	MCR	3	

Shutdown Cooling Flush
• Request Chemistry to verify Suppression Pool is within best practice limits of CSP-0006, Chemistry Surveillance and Scheduling System.
• IF suppression pool conductivity is NOT within best practice limits of CSP-0006, THEN perform a complete flush.
• IF desired and suppression pool conductivity is within best practice limits of CSP-0006, THEN perform the following:
  o Place RHR A(B) in suppression pool cooling.
  o Monitor E12-R610A(B), HX A(B) OUTLET CONDUCTIVITY OR Chemistry sample from SST-PNL80, and continue flush until conductivity is less than 2 umho/cm.
	MCR	3	Not required for plant shutdown or cooldown

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
○ Vent RHR A(B) HX as follows: ▪ Open E12-F074A(B), RHR A(B) HX UP STREAM VENT VALVE. ▪ Open E12-F073A(B), RHR A(B) HX DN STREAM VENT VALVE. ▪ WHEN at least 1 minute has elapsed, THEN close the following: • E12-F073A(B) • E12-F074A(B) • WHEN conductivity is less than 2 umho/cm, THEN perform the following: ○ Secure RHR A(B) from suppression pool cooling. ○ Close E12-VF085A(B), LPCS FILL PUMP STOP CHECK TO RHR A DISCH (DISCH FILL PUMP STOP CHECK TO RHR B DISCH). ○ Continue with pump warm-up as desired.			
During cooldown, review Attachment 7, High Critical Non-Safety Related MOVs That Are Susceptible to Thermal Binding and carryout actions to prevent thermal binding or actions to unbind the valves if they are closed when the valve temperature is greater than 200°F.	MCR	3	
IF MSIVs have been closed for pressure control, THEN the following systems may be utilized as necessary to continue a cooldown at less than or equal to 100°F/hr: • RWCU system per SOP-0090, Reactor Water Cleanup System (i.e. RWCU Blowdown Operation) • Main Steam Line Drains	MCR	3	
IF/WHEN reactor pressure is less than 400 psig, THEN any running reactor feedwater pumps may be shutdown per SOP-0009, Reactor Feedwater System.	MCR / TB 67'	3	Not required for plant shutdown or cooldown
Prior to reaching 135 psig, initiate monitoring of the following parameters: • RHR Room sump levels (monitor for possible reactor vessel inventory loss from shutdown cooling leakage). (DFR-LI135 and DFR-LI138). • Suppression Pool for unexpected level rise (monitor for reactor vessel inventory loss from RHR to the Suppression Pool).	MCR	3	
WHEN RPV pressure has been lowered to below 135 psig, THEN **place one loop of RHR in Shutdown Cooling per SOP-0031, Residual Heat Removal.**	MCR	3	
RHR Pump A Warmup (SOP-0031)			
Verify closed E12-F064A, RHR PUMP A MIN FLOW TO SUP PL.	MCR	3	

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Verify open E12-F047A, RHR A HX INLET VALVE.	MCR	3	
Verify closed E12-F004A, RHR PUMP A SUP PL SUCTION VALVE.	MCR	3	
Verify open E12-F006A, RHR PUMP A SDC SUCTION VALVE.	MCR	3	
On H13-P870, verify closed SPC-AOV16, SPC HX SW DISCH VLV.	MCR	3	
On H13-P870, throttle open E12-F068A, RHR HX A SVCE WTR RTN to establish less than or equal to 5800 gpm flow as indicated on E12-R602A, RHR HX A SVCE WTR FLOW.	MCR	3	
Close E12-F048A, RHR A HX BYPASS VALVE.	MCR	3	
Verify open E12-F003A, RHR A HX OUTLET VALVE.	MCR	3	
Verify open E12-F010, RHR SDC MAN ISOL VLV.	MCR	3	
In the Div 1 RSS Room at C61-PNL001, verify E12-MOVF008 ENABLE/DISABLE Switch is in ENABLE.	CB 95' Div 1 RSS Room	3	**REQUIRED**
Perform the following: • Depress B21H-S33, INBD ISOLATION SEAL-IN RESET Pushbutton. • Depress B21H-S32, OUTBD ISOLATION SEAL-IN RESET Pushbutton.	MCR	3	
At H13-P601, check RHR ISOLATION Status Lights are ON for E12-F008 and F009.	MCR	3	
Note current indicated CNS flow at LWS-PNL187 on CNS-FI116.	ACR	3	Not required for plant shutdown or cooldown
Slowly open E12-VF020, SHUTDOWN COOLING SUCTION FILL.	AB 95' RHR C Pump Room	3	Not required for plant shutdown or cooldown
WHEN CNS flow into the shutdown cooling header stops as indicated by a lack of flow noise or flow indication of approximately the same value as previously noted on CNS-FI116, THEN close E12-VF020.	AB 95' RHR C Pump Room	3	Not required for plant shutdown or cooldown
Open E12-F008, RHR SHUTDOWN COOLING OUTBD ISOL VALVE.	MCR	3	
Open E12-F009, RHR SHUTDOWN COOLING INBD ISOL VALVE.	MCR	3	
Perform STP-204-0204, RHR Shutdown Cooling Piping Fill Verification.	Steam Tunnel 114'	3	Not required for plant shutdown or cooldown
Notify Radwaste of reactor water flush to the Waste Collector Tanks.	MCR	3	

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Open E12-F049, RHR A TO RADWASTE UP STREAM ISOL VALVE.	MCR	3	
Throttle open E12-F040, RHR A TO RADWASTE DN STREAM ISOL VALVE.	MCR	3	
Monitor E12-R601, RHR TEMPERATURES, Point 1, RHR INLET TO HX1 A-1 (E12-N004A) for temperature rise and throttle E12-F040 to maintain less than or equal to 100°F/hr heatup.	MCR	3	
Continue the warm-up until E12-R601 Point 1 is within 100°F of reactor water temperature.	MCR	3	
Close E12-F049, RHR A TO RADWASTE UP STREAM ISOL VALVE.	MCR	3	
Close E12-F040, RHR A TO RADWASTE DN STREAM ISOL VALVE.	MCR	3	
Open E12-F048A, RHR HX A BYPASS VALVE.	MCR	3	
RHR Pump B Warm-up (SOP-0031)			
Verify closed E12-F064B, RHR PUMP B MIN FLOW TO SUP PL.	MCR	3	
Verify open E12-F047B, RHR B HX INLET VALVE.	MCR	3	
Verify closed E12-F004B, RHR PUMP B SUP PL SUCTION VALVE.	MCR	3	
Verify open E12-F006B, RHR PUMP B SDC SUCTION VALVE.	MCR	3	
IF Standby Service Water is supplying service water loads, THEN on H13-P870, verify closed SPC-AOV16, SPC HX SW DISCH VLV.	MCR	3	
On H13-P870, throttle open E12-F068B, RHR HX B SVCE WTR RTN to establish less than or equal to 5800 gpm flow as indicated on E12-R602B, RHR HX B SVCE WTR FLOW.	MCR	3	
Verify open E12-F010, RHR SDC MAN ISOL VLV.	MCR	3	
Verify closed E12-F049, RHR A TO RADWASTE UP STREAM ISOL VALVE.	MCR	3	
Verify closed E12-F040, RHR A TO RADWASTE DN STREAM ISOL VALVE.	MCR	3	
In the Div 1 RSS Room at C61-PNL001, verify E12-MOVF008 ENABLE/DISABLE Switch is in ENABLE.	CB 95' Div 1 RSS Room	3	**REQUIRED**
Perform the following: • Depress B21H-S32, OUTBD ISOLATION SEAL-IN RESET Pushbutton. • Depress B21H-S33, INBD ISOLATION SEAL-IN RESET Pushbutton.	MCR	3	
At H13-P601, check RHR ISOLATION Status Lights are ON for E12-F008 and F009.	MCR	3	
Note current indicated CNS flow at LWS-PNL187 on CNS-FI116.	ACR	3	Not required for plant shutdown or

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
			cooldown
Slowly open E12-VF020, SHUTDOWN COOLING SUCTION FILL.	AB 95' RHR C Pump Room	3	Not required for plant shutdown or cooldown
WHEN CNS flow into the shutdown cooling header stops as indicated by a lack of flow noise or flow indication of approximately the same value as previously noted on CNS-FI116, THEN close E12-VF020.	AB 95' RHR C Pump Room	3	Not required for plant shutdown or cooldown
Open E12-F008, RHR SHUTDOWN COOLING OUTBD ISOL VALVE.	MCR	3	
Open E12-F009, RHR SHUTDOWN COOLING INBD ISOL VALVE.	MCR	3	
Perform STP-204-0204, RHR Shutdown Cooling Piping Fill Verification.	Steam Tunnel 114'	3	Not required for plant shutdown or cooldown
Notify Radwaste of reactor water flush to the Waste Collector Tanks.	MCR	3	
Unlock and open E12-VF072B, RHR B DISCH LINE FLUSH.	AB 70' RHR B Pump Room	3	**REQUIRED**
Unlock and throttle open E12-VF070, RHR DR TO RADWASTE.	AB 80' RHR A Pump Room	3	**REQUIRED**
Monitor E12-R601, RHR TEMPERATURES, Point 11, RHR DISCH TO RADWASTE (E12-N024) and throttle E12-VF070 to maintain less than or equal to 100˚F/hr heatup.	MCR	3	
Continue the warm-up until E12-R601 Point 11 is within 100˚F of reactor water temperature.	MCR	3	
Close and lock E12-VF070, RHR DR TO RADWASTE.	AB 80' RHR A Pump Room	3	**REQUIRED**
Close and lock E12-VF072B, RHR B DISCH LINE FLUSH.	AB 70' RHR B Pump Room	3	**REQUIRED**
Startup of Shutdown Cooling (SOP-0031)			
IF any of the following RHR Shutdown Cooling interlocks are to be bypassed, THEN obtain senior plant management review and approval and verify contingency methods are in place to supply sufficient makeup water if a draining event occurs while the SDC interlocks are bypassed: • Low reactor water level isolation of E12-F008, RHR SHUTDOWN COOLING OUTBD ISOL VALVE and E12-F009, RHR SHUTDOWN COOLING INBD ISOL VALVE.	MCR	3	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
• Interlocks between E12-F004, RHR PUMP SUP PL SUCTION VALVE and E12-F006, RHR PUMP SDC SUCTION.			
On H13-P601, verify less than 135 psig Reactor Pressure as indicated on B21-R623A(B), RX LEVEL/PRESSURE RECORDER A(B).	MCR	3	
Verify closed the following: • E12-F004A(B), RHR PUMP A(B) SUP PL SUCTION VALVE • E12-F064A(B), RHR PUMP A(B) MIN FLOW TO SUP PL • E12-F024A(B), RHR PUMP A(B) TEST RTN TO SUP PL • E12-F037A(B), RHR A(B) TO UPPER POOL FPC ASSIST • E12-F048A(B), RHR A(B) HX BYPASS VALVE. • E12-F011A(B), RHR A(B) HX CNDS FLUSH TO SUP PL.	MCR	3	
Place in OFF and initiate administrative controls for EHS-MCC2E(2F) BKR 5C(7B), C002A(B) DISCH MIN FLOW VALVE.	AB 114' West	3	**REQUIRED**
IF Standby Service Water is supplying service water loads, THEN on H13-P870, verify closed SPC-AOV16, SPC HX SW DISCH VLV.	MCR	3	
On H13-P870, throttle open E12-F068A(B), RHR HX A(B) SVCE WTR RTN to establish less than or equal to 5800 gpm flow as indicated on H13-P601, E12-R602A(B), RHR HX A(B) SVCE WTR FLOW.	MCR	3	
Verify Step 4.4.2 has been performed.	MCR	3	
At H13-P601, depress B21H-S32, OUTBD ISOLATION SEAL-IN RESET Pushbutton.	MCR	3	
At H13-P601, depress B21H-S33, INBD ISOLATION SEAL-IN RESET Pushbutton.	MCR	3	
At H13-P601, check RHR ISOLATION Status Lights are ON for E12-F008 and E12-F009.	MCR	3	
In the Div 1 RSS Room at C61-PNL001, verify E12-MOVF008 ENABLE/DISABLE Switch is in ENABLE.	CB 95' Div 1 RSS Room	3	**REQUIRED**
Verify open the following: • E12-F010, RHR SDC MAN ISOL VLV • E12-F009, RHR SHUTDOWN COOLING INBD ISOL VALVE • E12-F008, RHR SHUTDOWN COOLING OUTBD ISOL VALVE • E12-F006A(B), RHR PUMP A(B) SDC SUCTION VALVE • E12-F047A(B), RHR A(B) HX INLET VALVE	MCR	3	
Verify open one of the following: • E12-F053A(B), RHR PUMP A(B) SDC INJECTION VALVE • E12-F037A(B), RHR A(B) TO UPPER POOL FPC ASSIST	MCR	3	
Close E12-F003A(B), RHR A(B) HX OUTLET VALVE.	MCR	3	
Start E12-C002A(B), RHR PUMP A(B) and IMMEDIATELY throttle open E12-F048A(B), RHR A(B) HX BYPASS VALVE to obtain greater	MCR	3	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
than or equal to 2000 gpm and less than or equal to 3000 gpm.			
Establish a stable flow of greater than or equal to 4000 gpm and less than or equal to 5000 gpm by throttling E12-F048A(B), RHR A(B) HX BYPASS VALVE.	MCR	3	
Throttle open E12-F003A(B), RHR A(B) HX OUTLET VALVE to approximately 10 PERCENT as indicated on E12-R611A(B), HX A(B) OUTLET VLV POS.	MCR	3	
Establish a cooldown rate of less than 100°F/hr as follows: • Slowly throttle open E12-F003A(B) RHR A(B) HX OUTLET VALVE and monitor the cooldown rate. • Throttle E12-F003A(B), RHR A(B) HX OUTLET VALVE and E12-F048A(B), RHR A(B) HX BYPASS VALVE to obtain the desired cooldown rate or maintain the desired coolant temperature while maintaining a constant RHR loop flow. • IF shifting divisions of Shutdown Cooling per Section 5.6, THEN in the other RHR loop, throttle E12-F003B(A), RHR B(A) HX OUTLET VALVE and E12-F048B(A), RHR B(A) HX BYPASS VALVE to maintain the desired cooldown rate or coolant temperature while maintaining a constant RHR loop flow. • Close FWS-MOV7A(B), A(B) FW OUTBD ISOL.	MCR	3	
IF it is desired to establish RHR Shutdown Cooling Protection, THEN Go To Section 4.5.	MCR / AB 95' & 115'		Not required for plant shutdown or cooldown
WHEN RHR Shutdown Cooling is established and adequate RPV makeup is assured via CRD or Feedwater, THEN close FWS-MOV7A(B), A(B) FW OUTBD ISOL valve on the Feedwater Header supporting RHR Shutdown Cooling.	MCR	3	
WHEN RPV cooldown is being conducted using RHR Shutdown Cooling, THEN stop discharging steam to the main condenser, break condenser vacuum and continue to shutdown the turbine plant as follows: • Place CONDENSER LOW VACUUM BYPASS Switches to BYPASS. • Close all turbine bypass valves and steam drain valves. • Open CNM-AOVVB, CNDS VAC BRKR. • WHEN condenser vacuum reaches approximately 0" Hg, THEN shutdown steam seals per SOP-0015, Gland Seal System and Exhaust System.	MCR	3	
WHEN condenser vacuum reaches 0" Hg AND mechanical vacuum pump operation is no longer required, THEN align Alternate Hotwell Level tygon tubing per SOP-0008, Condensate Storage, Makeup and Transfer.	TB 67'	3	Not required for plant shutdown or cooldown

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
At less than 190°F, perform the following: • Open B21-MOVF001, RX DN STREAM HEAD VENT TO DW EQPT DR SUMP. • Open B21-MOVF002, RX UP STREAM HEAD VENT TO DW EQPT DR SUMP. • Close B21-MOVF005, RX HEAD VENT TO MSL A. • Establish administrative controls to maintain the established vessel vent path until vessel head piping is disassembled.	MCR	3	
IF required/desired, THEN close the MSIVs by placing the following in CLOSE: • B21-F028B, MSL B OUTBD MSIV • B21-F028D, MSL D OUTBD MSIV • B21-F028A, MSL A OUTBD MSIV • B21-F028C, MSL C OUTBD MSIV • B21-F022B, MSL B INBD MSIV • B21-F022D, MSL D INBD MSIV • B21-F022A, MSL A INBD MSIV • B21-F022C, MSL C INBD MSIV	MCR	3	
At less than 200°F, Mode 4, perform the following: At H13-P632, place the following switches to BYPASS: • E31A-S1A, RWCU ISOLATION BYPASS DIV 1 • E31A-S2A, RCIC ISOLATION BYPASS DIV 1 • E31A-S4A, RHR ISOLATION BYPASS DIV 1	MCR	3	
At H13-P642, place the following switches to BYPASS: • E31A-S1B, RWCU ISOLATION BYPASS DIV 2 • E31A-S2B, RCIC ISOLATION BYPASS DIV 2 • E31A-S4B, RHR ISOLATION BYPASS DIV 2	MCR	3	
Implement Shutdown Cooling Protection per SOP-0031, Residual Heat Removal.	MCR / AB 95' & 115'	3	Not required for plant shutdown or cooldown
Bypass RPS trip logic using EOP-0005 Enclosure 12 Bypass Switches per SOP-0079, Reactor Protective System.	MCR	3	
Bypass ARI logic trips per SOP-0079.	MCR	3	
Bypass Backup Scram Valve trips per SOP-0079.	MCR	3	
At less than 200°F, Mode 4, perform the following to prevent isolating Breathing Air: • Verify open SAS-MOV102, SVCE AIR OUTBD ISOL	MCR	3	Not required for plant shutdown or

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
			cooldown
• Open EHS-MCC2J Bkr 3C, SAS-MOV102 CONTAINMENT SERVICE AIR ISOLATION.	AB 141' West	3	Not required for plant shutdown or cooldown
• Hang the following SAS-MOV102 Caution Tags stating that before closing the breaker or valve, verify that Breathing Air is not in use.: o EHS-MCC2J Bkr 3C o SAS-MOV102 local handwheel o SAS-MOV102 MCR control switch	MCR	3	Not required for plant shutdown or cooldown
• Hang "Breathing Air in Use" sign in the MCR.	MCR	3	Not required for plant shutdown or cooldown
• Make an "Open Item" Narrative Log entry by checking the "Open Item" box stating that "Breathing Air Is in Use" to carry over until Breathing Air is no longer in use.	MCR	3	Not required for plant shutdown or cooldown
At less than 200°F, notify Chemistry to consider securing the Durability Monitor.	AB 114' Crescent Area	3	Not required for plant shutdown or cooldown
IF required/desired, THEN shutdown Reactor Recirculation System per SOP-0003, Reactor Recirculation System and raise reactor water level to at least 75 inches on shutdown range level instrumentation.	MCR	3	Not required for plant shutdown or cooldown
As necessary, reduce the number of operating Turbine Building Chillers per SOP-0064 to prevent the chillers from tripping on low load.	TB 67'	3	Not required for plant shutdown or cooldown
GOP-0003 Scram Recovery			
Verify/establish on-scale neutron monitoring on the SRMs and IRMs • VERIFY the SRM Channel Functional Tests are current. If Channel Functional Tests are not current, refer to Tech Spec 3.3.1.2.	MCR	3	
Maintain RPV pressure to prevent excessive cooldown rates or RPV overpressurization by: • Use of Main Turbine Bypass System. • Use of Normal Plant Steam Loads/Steam Line Drains. • RCIC in CST to CST Mode per SOP-0035 Reactor Core Isolation Cooling System.	MCR	3	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
• Alternate opening of SRVs as needed (only on RPV isolation).			
Maintain RPV Level using: • Condensate/Feedwater • RCIC • HPCS • CRD/RWCU	MCR	3	
If tripped, restart Reactor Recirculation Pumps on LFMG per SOP-0003, Reactor Recirculation: • Open flow control valves to the full open position.	MCR / LFMG Room	3	Not required for plant shutdown or cooldown
Verify Main Turbine steam seals are being maintained at approximately 4 psig.	MCR	3	
Start air removal pumps per SOP-0025 Condenser Air Removal System. • Maintain condenser vacuum between 23" Hg and 28" Hg.	MCR / TB 67'	3	Not required for plant shutdown or cooldown
IF the Steam Jet Air Ejectors have been lost, THEN Secure Offgas System per SOP-0092, to establish purge air flow in order to prevent system reverse flow.	MCR / TB 95' & 123'	3	Not required for plant shutdown or cooldown
Notify Chemistry Department to operate the Offgas Hydrogen Analyzers per COP-0227, Operation of the Offgas Hydrogen Analyzers.	MCR / TB 123'	3	Not required for plant shutdown or cooldown
Record the highest vessel pressure indicated by tracking pointer on B21-PIR004A and B21-PIR004B (114' Containment). Reset the tracking pointer.	RB 114'	3	Not required for plant shutdown or cooldown
Inspect all CRD HCUs for leakage due to piping cracks. • Notify Engineering NDE that visual inspections of HCU charging water piping are required.	RB 114'	3	Not required for plant shutdown or cooldown
WHEN FWREG Valves are removed from service, THEN perform SOP-0009 Attachment for Calibration Check of FWREG Valve.	MCR / TB 67'	3	Not required for plant shutdown or cooldown
Go To GOP-0002 - Start at the beginning of GOP-0002 and complete	MCR	3	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
all steps required to place the plant in the desired mode after scram.			
IF the plant tripped while connected to the grid, THEN notify Site Design Engineering to notify the TOP personnel of the event per ENS-DC-201, ENS Transmission Grid Monitoring Attachment 9.3 Step 3.0[1].	MCR	3	
Engineering perform a review of post scram cooldown data and compare to PT Curves provided in STP-050-0700 Attachment 3. Also verify the cooldown rate is bounded by analyzed thermal cycles.	MCR	3	
Shift Manager to perform a Post SCRAM crew critique identifying all Human Performance issues and Equipment Malfunctions. Document each item on separate CRs. Attach Crew Critique to this procedure.	MCR	3	
AOM/Shift Manager review equipment malfunctions and recommend to OSRC required repairs prior to restart. This should include an evaluation of risk mitigating Non TRM structures, systems, and components.	MCR	3	

Control Room ventilation systems have adequate engineered safety/design features in place to preclude a Control Room evacuation due to the release of a hazardous gas. Therefore, the Control Room is not included in this assessment or in Table H-2.

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

Table A-3 & H-2 Results

Table A-3 & H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode**
Auxiliary Building 70' RHR B Pump Room	3
Auxiliary Building 80' RHR A Pump Room	3
Auxiliary Building 114' West	3
Control Building 95' Div 1 RSS Room	3

Mode 3 is included above for SDC-related activities because the procedures begin alignment in Mode 3; however, these actions could be delayed until Mode 4, if necessary. In order to ensure adequate guidance to emergency response personnel, the above areas are added to the EAL in order to provide prompt operator guidance for EAL declaration.

Control Room ventilation systems have adequate engineered safety/design features in place to preclude a Control Room evacuation due to the release of a hazardous gas. Therefore, the Control Room is not included in this assessment or in Table H-2.

ENCLOSURE 3
RBG-47847

PROPOSED EAL TECHNICAL BASIS DOCUMENT (CLEAN)

River Bend Station
EAL Technical Basis

Table of Contents

1.0 INTRODUCTION

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for River Bend Station (RBS). It should be used to facilitate review of the RBS EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EIP-2-001, Classification of Emergencies, 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 to be made as soon as conditions are present and recognizable for the classification, but within 15 minutes or less in all cases of conditions present. Use of this document for assistance is not intended to delay the emergency classification.

Because the information in a basis document can affect emergency classification decision-making (e.g., the Emergency Director 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 RBS Emergency Plan.

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

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

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

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

2.2 Fission Product Barriers

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

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

The primary fission product barriers are:

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

B. Reactor Coolant System Barrier (RCB): The RCS Barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping out to and including the isolation valves.

C. Containment Barrier (CNB): The Primary Containment Barrier includes the drywell, the containment, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. Containment Barrier thresholds are used as criteria for escalation of the Emergency Classification Level (ECL) from Alert to a Site Area Emergency or a General Emergency.

2.3 Fission Product Barrier Classification Criteria

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

Alert:

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

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

2.4 EAL Organization

The RBS EAL scheme includes the following features:

- Division of the EAL set into three broad groups:

- EALs applicable under <u>any</u> plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
- EALs applicable only under <u>hot</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Startup, or Power Operation mode.
- EALs applicable only under <u>cold</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling or Defueled mode.

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

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

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

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

EAL Groups, Categories and Subcategories

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

2.5 Technical Bases Information

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

Category Letter & Title

Subcategory Number & Title

Initiating Condition (IC)

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

EAL Identifier (enclosed in rectangle)

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

1. First character (letter): Corresponds to the EAL category as described above (A, C, E, F, H or S)
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 Shutdown, 4 - Cold Shutdown, 5 - Refueling, DEF - Defueled, or All. (See Section 2.6 for operating mode definitions)

Definitions:

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

Basis:

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

Reference(s):

Source documentation from which the EAL is derived

2.6 Operating Mode Applicability

1 Power Operation

Reactor is critical and the mode switch is in RUN

2 Startup

The mode switch is in REFUEL (with all reactor vessel head closure bolts fully tensioned) or STARTUP/HOT STANDBY

3 Hot Shutdown

The mode switch is in SHUTDOWN and average reactor coolant temperature is >200°F

4 Cold Shutdown

The mode switch is in SHUTDOWN and average reactor coolant temperature is ≤ 200°F

5 Refueling

The mode switch is in REFUEL or SHUTDOWN with one or more reactor vessel head closure bolts are less than fully tensioned

DEF Defueled

RPV contains no irradiated fuel

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

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the 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 informing basis information. In the Recognition Category F matrices, EALs are based on loss or potential loss of Fission Product Barrier Thresholds.

EAL matrices should be read from left to right, from General Emergency to Unusual Event, and top to bottom. Declaration decisions should be independently verified before declaration is made except when gaining this verification would exceed the 15 minute declaration requirement. Place keeping should be used on all EAL matrices.

3.1.1 Classification Timeliness

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

3.1.2 Valid Indications

All emergency classification assessments shall be based upon valid indications, reports or conditions. A valid indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy. 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 to be valid when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

3.1.3 Imminent Conditions

For ICs and EALs that have a stipulated time duration (e.g., 15 minutes, 30 minutes, etc.), the Emergency 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).

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.

3.2 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. 4.1.8).

3.2.1 Classification of Multiple Events and Conditions

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

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

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

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

3.2.2 Consideration of Mode Changes During Classification

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

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

3.2.3 Classification of Imminent Conditions

Although EALs provide specific thresholds, the Emergency 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 Termination

An ECL may be terminated when the event or condition that meets the classified IC and EAL no longer exists, and other site-specific termination requirements are met.

3.2.5 Classification of Short-Lived Events

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

3.2.6 Classification of Transient Conditions

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

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

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

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

It is important to stress that the 15-minute emergency classification assessment period (process clock) is not a "grace period" during which a classification may be delayed to allow the performance of a corrective action that would obviate the need to classify the event. Emergency classification assessments must be deliberate and timely, with no undue delays. The provision discussed above addresses only those rapidly evolving situations when an operator is able to take a successful corrective action prior to the Emergency 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 ML12326A805

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

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

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

4.1.5 10 § CFR 50.73 License Event Report System

4.1.6 RBS Technical Specifications Table 1.1-1, Modes

4.1.7 RBS USAR Section 2.1 Geography and Demography

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

4.1.9 RBS Emergency Plan

4.1.10 RBS USAR 9.1.2.5 Holtec HI-STORM Dry Spent Fuel Storage System

4.1.11 RBS USAR 9.1.4.2.3.11 Fuel Transfer System

4.1.12 OSP-0037 Shutdown Operations Protection Plan (SOPP)

4.1.13 RBS Security Plan

4.2 Implementing

4.2.1 EIP-2-001 Classification of Emergencies

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

4.2.3 RBS EAL Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

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

Alert

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

Confinement Boundary

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the RBS ISFSI, Confinement Boundary is defined as the Holtec System Multi-Purpose Canister (MPC) (ref. 4.1.10).

Containment Closure

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

Containment Closure is established when the containment requirements of OSP-0037 (ref. 4.1.12) are met and at least one integral barrier to the release of radioactive material is provided, within the specified limits using STP-057-3804.

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)

Explosion

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

Fire

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

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 RBS 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 RBS. 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.

Owner Controlled Area (OCA)

For the purposes of classification, the Security Owner Controlled Area (SOCA) or the area between the SOCA Fence and the PROTECTED AREA Boundary (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

The area within the perimeter of the RBS security fence. (ref. 4.1.9).

Refueling Pathway

Reactor cavity (well), containment spent fuel pool, fuel transfer canal, and fuel building spent fuel pools, but **not** including the reactor vessel, comprise the refueling pathway (ref. 4.1.11).

Restore

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

Safety System

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

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

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

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

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

Security Condition

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

Site Area Emergency

Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE 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

For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline. (ref. 4.1.7)

Unisolable

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

Unplanned

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

Unusual Event

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

Valid

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

Visible Damage

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

5.2 Abbreviations/Acronyms

°F .... Degrees Fahrenheit
° .... Degrees
AB .... Auxiliary Building
AC .... Alternating Current
AOP .... Abnormal Operating Procedure
APRM .... Average Power Range Meter
ATWS .... Anticipated Transient Without Scram
BWR .... Boiling Water Reactor
BWROG .... Boiling Water Reactor Owners Group
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
CNB .... Containment Barrier
CS .... Core Spray
CTMT .... Containment
DEF .... Defueled
DBA .... Design Basis Accident
DC .... Direct Current
D/G .... Diesel Generator
DRMS .... Digital Radiation Monitoring System
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
EPG .... Emergency Procedure Guideline
EPP .... Emergency Plan Procedure
ERO .... Emergency Response Organization
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
HCTL .... Heat Capacity Temperature Limit

HPCS....................High Pressure Core Spray
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
LPCS....................Low Pressure Core Spray
LRW....................Liquid Radwaste
LWR....................Light Water Reactor
MPC....................Maximum Permissible Concentration/Multi-Purpose Canister
MPH....................Miles Per Hour
mR, mRem, mrem, mREM ....milli-Roentgen Equivalent Man
MSCRWL....................Minimum Steam Colling RPV Water Level
MSIV....................Main Steam Isolation Valve
MSL....................Main Steam Line
MW....................Megawatt
NEI....................Nuclear Energy Institute
NEIC....................National Earthquake Information Center
NESP....................National Environmental Studies Project
NORAD....................North American Aerospace Defense Command
(NO)UE....................Notification of Unusual Event
NPP....................Nuclear Power Plant
NRC....................Nuclear Regulatory Commission
NSSS....................Nuclear Steam Supply System
OBE....................Operating Basis Earthquake
ODCM....................Offsite Dose Calculation Manual
ORO....................Offsite Response Organization
PA....................Protected Area
PAG....................Protective Action Guideline
PB....................Pushbutton
PCIS....................Primary Containment Isolation System
PRA/PSA....................Probabilistic Risk Assessment / Probabilistic Safety Assessment
PSID....................Pounds Per Square Inch Differential
PSIG....................Pounds per Square Inch Gauge

R	Roentgen
RCB	RCS Barrier
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
Rem, rem, REM	Roentgen Equivalent Man
RETS	Radiological Effluent Technical Specifications
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup
SAG	Severe Accident Guideline
SAP	Severe Accident Procedure
SAR	Safety Analysis Report
SBO	Station Blackout
SCBA	Self-Contained Breathing Apparatus
SOCA	Security Owner Controlled Area
SPDS	Safety Parameter Display System
SRO	Senior Reactor Operator
SRV	Safety Relief Valve
SSE	Safe Shutdown Earthquake
TEDE	Total Effective Dose Equivalent
TAF	Top of Active Fuel
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report
USGS	United States Geological Survey

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

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

RBS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
AU1.1	AU1	1, 2
AU1.2	AU1	3
AU2.1	AU2	1
AA1.1	AA1	1
AA1.2	AA1	2
AA1.3	AA1	3
AA1.4	AA1	4
AA2.1	AA2	1
AA2.2	AA2	2
AA2.3	AA2	3
AA3.1	AA3	1
AA3.2	AA3	2
AS1.1	AS1	1
AS1.2	AS1	2
AS1.3	AS1	3
AS2.1	AS2	1
AG1.1	AG1	1
AG1.2	AG1	2
AG1.3	AG1	3

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

RBS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
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, 2
HA5.1	HA5	1
HA6.1	HA6	1
HA7.1	HA7	1
HS1.1	HS1	1
HS6.1	HS6	1
HS7.1	HS7	1
HG7.1	HG7	1
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	1
SU4.2	SU3	2
SU5.1	SU4	1, 2, 3

RBS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
SU6.1	SU5	1
SU6.2	SU5	2
SU7.1	SU6	1, 2, 3
N/A	SU7	1, 2
SA1.1	SA1	1
SA3.1	SA2	1
SA6.1	SA5	1
SA8.1	SA9	1
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1
SG1.1	SG1	1
SG1.2	SG8	1

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

Category A – Abnormal Rad Levels / 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: A – 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:

AU1.1 Unusual Event

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

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

Table A-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	Main Plant Vent - Primary	RE125	9.56E+08 µCi/sec	9.56E+07 µCi/sec	9.63E+06 µCi/sec	1.01E+05 µCi/sec
Gaseous	Main Plant Vent - Secondary	RE126	----	----	1.66E-01 µCi/ml	1.74E-03 µCi/ml
Gaseous	Fuel Bldg Vent - Primary	RE5A	7.75E+08 µCi/sec	7.75E+07 µCi/sec	7.75E+06 µCi/sec	6.50E+03 µCi/sec
Gaseous	Fuel Bldg Vent - Secondary	RE5B	----	----	1.72E-01µCi/ml	1.38E-03 µCi/ml
Gaseous	Radwaste Bldg Vent - Primary	RE6A	8.03E+08 µCi/sec	8.03E+07 µCi/sec	8.03E+06 µCi/sec	6.96E+04 µCi/sec
Gaseous	Radwaste Bldg Vent - Secondary	RE6B	----	----	2.12E-01 µCi/ml	1.71E-04 µCi/ml
Liquid	Liquid Radwaste	RE107	----	----	----	2 x Alarm Setpoint

Mode Applicability:

All

Definition(s):

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

Basis:

This IC addresses a potential reduction 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 as well as radioactivity releases that cause effluent radiation monitor readings to exceed 2 times the limit established by a radioactivity discharge permit. Such releases are typically 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 AA1.

Reference(s):

1. SOP-0086 Digital Radiation Monitoring System
2. RSP-0008 Offsite Dose Calculation Manual
3. EP-CALC-RBS-1801 Radiological Effluent EAL Threshold Values
4. NEI 99-01 AU1

Category: A – 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:

AU1.2	**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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

None

Basis:

This IC addresses a potential reduction 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 AA1.

Reference(s):

1. RSP-0008 Offsite Dose Calculation Manual
2. NEI 99-01 AU1

Category: A – 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:

AA1.1 Alert

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

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

Note 4 The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table A-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	Main Plant Vent - Primary	RE125	9.56E+08 μCi/sec	9.56E+07 μCi/sec	9.63E+06 μCi/sec	1.01E+05 μCi/sec
	Main Plant Vent - Secondary	RE126	----	----	1.66E-01 μCi/ml	1.74E-03 μCi/ml
	Fuel Bldg Vent - Primary	RE5A	7.75E+08 μCi/sec	7.75E+07 μCi/sec	7.75E+06 μCi/sec	6.50E+03 μCi/sec
	Fuel Bldg Vent - Secondary	RE5B	----	----	1.72E-01μCi/ml	1.38E-03 μCi/ml
	Radwaste Bldg Vent - Primary	RE6A	8.03E+08 μCi/sec	8.03E+07 μCi/sec	8.03E+06 μCi/sec	6.96E+04 μCi/sec
	Radwaste Bldg Vent - Secondary	RE6B	----	----	2.12E-01 μCi/ml	1.71E-04 μCi/ml
Liquid	Liquid Radwaste	RE107	----	----	----	2 x Alarm Setpoint

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem 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.

Reference(s):

1. SOP-0086 Digital Radiation Monitoring System
2. RSP-0008 Offsite Dose Calculation Manual
3. EP-CALC-RBS-1801 Radiological Effluent EAL Threshold Values
4. NEI 99-01 AA1

Category: A – 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:

AA1.2 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 AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline.

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.

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

Reference(s):

1. EIP-2-024 Offsite Dose Calculations
2. NEI 99-01 AA1

Category: A – 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:

AA1.3 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline.

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.

This EAL is assessed per the ODCM (ref. 2)

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

Reference(s):

1. EIP-2-024 Offsite Dose Calculations
2. RSP-0008 Offsite Dose Calculation Manual
3. NEI 99-01 AA1

Category: A – 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:

AA1.4 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline.

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.

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

Reference(s):

1. EIP-2-014 Offsite Radiological Monitoring
2. NEI 99-01 AA1

Category: A – 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:

AS1.1 Site Area Emergency

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

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

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table A-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Main Plant Vent - Primary	RE125	9.56E+08 μCi/sec	9.56E+07 μCi/sec	9.63E+06 μCi/sec	1.01E+05 μCi/sec
Gaseous	Main Plant Vent - Secondary	RE126	----	----	1.66E-01 μCi/ml	1.74E-03 μCi/ml
Gaseous	Fuel Bldg Vent - Primary	RE5A	7.75E+08 μCi/sec	7.75E+07 μCi/sec	7.75E+06 μCi/sec	6.50E+03 μCi/sec
Gaseous	Fuel Bldg Vent - Secondary	RE5B	----	----	1.72E-01μCi/ml	1.38E-03 μCi/ml
Gaseous	Radwaste Bldg Vent - Primary	RE6A	8.03E+08 μCi/sec	8.03E+07 μCi/sec	8.03E+06 μCi/sec	6.96E+04 μCi/sec
Gaseous	Radwaste Bldg Vent - Secondary	RE6B	----	----	2.12E-01 μCi/ml	1.71E-04 μCi/ml
Liquid	Liquid Radwaste	RE107	----	----	----	2 x Alarm Setpoint

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

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

Reference(s):

1. SOP-0086 Digital Radiation Monitoring System
2. RSP-0008 Offsite Dose Calculation Manual
3. EP-CALC-RBS-1801 Radiological Effluent EAL Threshold Values
4. NEI 99-01 AS1

Category: A – 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:

AS1.2	**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 AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline.

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 AG1.

Reference(s):

1. EIP-2-024 Offsite Dose Calculations
2. NEI 99-01 AS1

Category: A – 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:

AS1.3 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline.

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 AG1.

Reference(s):

1. EIP-2-014 Offsite Radiological Monitoring
2. NEI 99-01 AS1

Category: A – 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:

AG1.1 General Emergency

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

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

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table A-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Main Plant Vent - Primary	RE125	9.56E+08 µCi/sec	9.56E+07 µCi/sec	9.63E+06 µCi/sec	1.01E+05 µCi/sec
Gaseous	Main Plant Vent - Secondary	RE126	----	----	1.66E-01 µCi/ml	1.74E-03 µCi/ml
Gaseous	Fuel Bldg Vent - Primary	RE5A	7.75E+08 µCi/sec	7.75E+07 µCi/sec	7.75E+06 µCi/sec	6.50E+03 µCi/sec
Gaseous	Fuel Bldg Vent - Secondary	RE5B	----	----	1.72E-01µCi/ml	1.38E-03 µCi/ml
Gaseous	Radwaste Bldg Vent - Primary	RE6A	8.03E+08 µCi/sec	8.03E+07 µCi/sec	8.03E+06 µCi/sec	6.96E+04 µCi/sec
Gaseous	Radwaste Bldg Vent - Secondary	RE6B	----	----	2.12E-01 µCi/ml	1.71E-04 µCi/ml
Liquid	Liquid Radwaste	RE107	----	----	----	2 x Alarm Setpoint

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

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

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

Reference(s):

1. SOP-0086 Digital Radiation Monitoring System
2. RSP-0008 Offsite Dose Calculation Manual
3. EP-CALC-RBS-1801 Radiological Effluent EAL Threshold Values
4. NEI 99-01 AG1

Category: A – 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:

AG1.2 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 AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline.

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.

Reference(s):

1. EIP-2-024 Offsite Dose Calculations
2. NEI 99-01 AG1

Category: A – 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:

AG1.3 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline

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.

Reference(s):

1. EIP-2-014 Offsite Radiological Monitoring
2. NEI 99-01 AG1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: UNPLANNED loss of water level above irradiated fuel

EAL:

AU2.1 Unusual Event

UNPLANNED water level drop in the REFUELING PATHWAY as indicated by level instrumentation, low water level alarm or visual observation.

AND

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

- RMS-RE140 Refueling Floor Near North Entrance
- RMS-RE141 Refueling Floor Near South Entrance
- RMS-RE192 Fuel Building Operating Floor - South
- RMS-RE193 Fuel Building Operating Floor - North

Mode Applicability:

All

Definition(s):

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

REFUELING PATHWAY- Reactor cavity (well), containment spent fuel pool, fuel transfer canal, and fuel building spent fuel pools, but **not** including the reactor vessel, comprise the refueling pathway.

Basis:

This IC addresses a drop 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 drop will be primarily determined by indications from available level instrumentation. Other sources of level indications may include reports from plant personnel (e.g., from a refueling crew) or video camera observations (if available). A significant drop in the water level may also cause a rise 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 rise 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.

The following low level alarms on P870 are related to low level in the REFUELING PATHWAY (ref. 4):

- SFP low alarm #0111 (H13-P870 / 56A / E02)
- Upper Transfer Pool Low alarm #0336 (H13-P870 / 56A / E03)
- Cask Pool Low alarm #0337 (H13-P870 / 56A / D03)
- Lower Transfer Pool low alarm #0335 (H13-P870 / 56A / F03)
- Rx Bldg Storage Pool low alarm #0112 (H13-P870 / 56A / H03)

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.

Reference(s):

1. AOP-0027 Fuel Handling Mishaps
2. USAR 12.3 Table 12.3-1 Area Direct Radiation Monitor Locations
3. USAR 9.1.4.2.3.11 Fuel Transfer System
4. ARP-870-0034 P870-56 Alarm Response
5. NEI 99-01 AU2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AA2.1 Alert

IMMINENT uncovery of irradiated fuel in the REFUELING PATHWAY

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY- The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the RBS ISFSI, Confinement Boundary is defined as the Holtec System Multi-Purpose Canister (MPC).

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

REFUELING PATHWAY- Reactor cavity (well), containment spent fuel pool, fuel transfer canal, and fuel building spent fuel pools, but **not** including the reactor vessel, comprise the refueling pathway.

Basis:

This IC addresses events that have caused IMMINENT or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the REFUELING PATHWAY. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant. 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 EU1.

This EAL escalates from AU2.1 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 a rise in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.

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

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

Reference(s):

1. NEI 99-01 AA2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity

AND

High alarm on **any** Table A-2 radiation monitor

Table A-2 Fuel Damage Radiation Monitors
• RMS-RE140 Refueling Floor Near North Entrance • RMS-RE141 Refueling Floor Near South Entrance • RMS-RE192 Fuel Building Operating Floor - South • RMS-RE193 Fuel Building Operating Floor - North • RMS-RE5A(B) Fuel Building Ventilation Exhaust

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY- The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the RBS ISFSI, the Confinement Boundary is comprised the Holtec System Multi-Purpose Canister (MPC).

Basis:

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

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

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

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

Reference(s):

1. AOP-0027 Fuel Handling Mishaps
2. USAR 12.3 Table 12.3-1 Area Direct Radiation Monitor Locations
3. USAR 12.3 Table 12.3-2 Airborne Process and Effluent Radiation Monitors
4. NEI 99-01 AA2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AA2.3 Alert

Lowering of spent fuel pool level to 108.0 ft. (Level 2) on SFC-LI29A/B

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

Escalation of the emergency classification level would be via IC AS1 or AS2.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (SFC-LI29A/B) capable of identifying normal level (Level 1), SFP level approximately 23 ft. above the top of the fuel racks, (Level 2) 107 ft. 10 5/16 in. (rounded to 108.0 ft.) which is that level adequate to provide substantial radiation shielding for a person standing on the SFP operating deck, and SFP level at the top of the fuel racks (Level 3) 85 ft. 10 5/16 in. (rounded to 86.0 ft.) (ref. 1). RBS uses a Level 3 of approximately one foot above the highest point of any fuel rack providing added margin.

Spent Fuel Pool Level indicators SFC-LI29A and B are read on the 98 ft. elevation Control Building on the interior of the West exterior wall (ref. 2).

Reference(s):

1. RBG-47570 Completion of Required Action by NRC Order EA-12-051 Reliable SFP Instrumentation
2. RBS-FSG-011 Alternate Spent Fuel Pool Makeup and Cooling
3. NEI 99-01 AA2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AS2.1	**Site Area Emergency**
Lowering of spent fuel pool level to 86.0 ft. (Level 3) on SFC-LI29A/B	

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

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

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (SFC-LI29A/B) capable of identifying normal level (Level 1), SFP level approximately 23 ft. above the top of the fuel racks, (Level 2) 107 ft. 10 5/16 in. (rounded to 108.0 ft.) which is that level adequate to provide substantial radiation shielding for a person standing on the SFP operating deck, and SFP level at the top of the fuel racks (Level 3) 85 ft. 10 5/16 in. (rounded to 86.0 ft.) (ref. 1). RBS uses a Level 3 of approximately one foot above the highest point of any fuel rack providing added margin.

Spent Fuel Pool Level indicators SFC-LI29A and B are read on the 98 ft. elevation Control Building on the interior of the West exterior wall (ref. 2).

Reference(s):

1. RBG-47570 Completion of Required Action by NRC Order EA-12-051 Reliable SFP Instrumentation
2. RBS-FSG-011 Alternate Spent Fuel Pool Makeup and Cooling
3. NEI 99-01 AS2

Category: A – 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:

AG2.1 General Emergency

Spent fuel pool level **cannot** be restored to at least 86.0 ft. (Level 3) on SFC-LI29A/B for ≥ 60 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (SFC-LI29A/B) capable of identifying normal level (Level 1), SFP level approximately 23 ft. above the top of the fuel racks, (Level 2) 107 ft. 10 5/16 in. (rounded to 108.0 ft.) which is that level adequate to provide substantial radiation shielding for a person standing on the SFP operating deck, and SFP level at the top of the fuel racks (Level 3) 85 ft. 10 5/16 in. (rounded to 86.0 ft.) (ref. 1). RBS uses a Level 3 of approximately one foot above the highest point of any fuel rack providing added margin.

Spent Fuel Pool Level indicators SFC-LI29A and B are read on the 98 ft. elevation Control Building on the interior of the West exterior wall (ref. 2).

Reference(s):

1. RBG-47570 Completion of Required Action by NRC Order EA-12-051 Reliable SFP Instrumentation
2. RBS-FSG-011 Alternate Spent Fuel Pool Makeup and Cooling
3. NEI 99-01 AG2

Category: A – 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:

AA3.1 Alert

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

- Control Room (RMS-RE170)
- Central Alarm Station (by survey)

Mode Applicability:

All

Definition(s):

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

Basis:

Areas that meet this threshold include the Control Room (CR) and the Central Alarm Station (CAS). The Control Room is monitored for excessive radiation by one detector, RMS-RE170 (ref. 1). The CAS is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations. There are no permanently installed area radiation monitors in CAS that may be used to assess this EAL threshold. Therefore, this threshold is evaluated using local radiation survey for this area.

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 rise in radiation levels and determine if another IC may be applicable.

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

Reference(s):

1. USAR 12.3 Table 12.3-1 Area Direct Radiation Monitor Locations
2. NEI 99-01 AA3

Category: A – 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:

AA3.2 Alert

An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table A-3 room or area (Note 5)

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

Table A-3 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode**
Auxiliary Building 70' RHR B Pump Room	3
Auxiliary Building 80' RHR A Pump Room	3
Auxiliary Building 114' West	3
Control Building 95' Div 1 RSS Room	3

Mode Applicability:

3 – Hot Shutdown

Definition(s):

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

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

Basis:

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

For AA3.2, 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 higher 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 rise occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 3.
- The higher 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 access control measures are of a conservative or precautionary nature, and would not actually prevent or IMPEDE a required action.

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

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).

EAL AA3.2 mode applicability has been limited to the mode limitations of Table A-3 (Mode 3 **only**).

Reference(s):

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

Category C – Cold Shutdown / Refueling System Malfunction

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

Category C EALs are directly associated with cold shutdown or refueling system safety functions. Given the variability of plant configurations (e.g., systems out-of-service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The cold shutdown and refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, CONTAINMENT CLOSURE, and fuel clad integrity for the applicable operating modes (4 - Cold Shutdown, 5 - Refueling, DEF – Defueled).

The events of this category pertain to the following subcategories:

1. RPV Level

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

2. Loss of Emergency AC Power

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

3. RCS Temperature

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

4. Loss of Vital DC Power

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

5. Loss of Communications

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

6. Hazardous Event Affecting Safety Systems

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

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: UNPLANNED loss of RPV inventory

EAL:

CU1.1 Unusual Event

UNPLANNED loss of reactor coolant results in RPV water 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

Basis:

River Bend is equipped with multiple RPV water level instruments including: Wide Range, Fuel Zone, Shutdown Range, Upset Range, and Narrow Range (ref. 1). Multiple instruments on different reference and variable legs should be monitored. The Upset Range and Shutdown Range instruments share a common reference leg; therefore, Narrow Range instruments should be routinely monitored when relying on Shutdown or Upset Range instrument as the primary indication.

With the plant in Cold Shutdown, RPV water level is normally maintained above the RPV low level scram setpoint of 9.7 in. (ref. 2). However, if RPV level is being controlled below the RPV low level scram 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, RPV water level is normally maintained at or above the reactor vessel flange. Technical Specifications require at least 23 ft of water above the top of the reactor vessel flange in the refueling cavity during refueling operations (ref. 3). The RPV flange is at approximately 200 in. on the Shutdown Range. (ref. 4).

This EAL 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 RPV level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that lower RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level lowering 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 RPV level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.

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

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

Reference(s):

1. EOP-1 RPV Control
2. Technical Requirements Table 3.3.1.1-1 RPS Instrumentation
3. Technical Specification 3.9.6 Reactor Pressure Vessel (RPV) Water Level - Irradiated Fuel
4. GMP-0102 Reactor Vessel Disassembly
5. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: UNPLANNED loss of RPV inventory

EAL:

CU1.2 Unusual Event

RPV water level **cannot** be monitored

AND EITHER

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

Table C-1 Sumps/Pool
• Drywell equipment drain sump
• Drywell floor drain sump
• Pedestal floor drain sump
• CTMT equipment drain sump
• CTMT floor drain sump
• Suppression Pool
• RHR A, B, C, HPCS, LPCS, RCIC room sumps
• Auxiliary Building floor drain sump

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

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

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

Basis:

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

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Rise in drywell equipment drain sump level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 1, 2). An Auxiliary Building sump level rise may also be indicative of RCS inventory losses external to the Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

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 RPV level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that lower RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level lowering 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 RPV level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels (Table C-1). Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RPV.

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

Reference(s):

1. SOP-0104 Floor and Equipment Drains System
2. SOP-0033 Drywell and Containment Leak Detection System
3. EOP-3 Secondary Containment and Radioactivity Release Control
4. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Significant Loss of RPV inventory

EAL:

CA1.1	**Alert**
Loss of RPV inventory as indicated by RPV water level < -43 in. (Level 2)	

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

None

Basis:

The threshold RPV water level of -43 in. is the Level 2 actuation setpoint for HPCS and RCIC. Although RCIC cannot restore RPV inventory in the cold condition, the Level 2 actuation setpoint is operationally significant and is indicative of a loss of RPV inventory significantly below the low RPV water level scram setpoint specified in CU1.1 (ref. 1, 2).

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

For this EAL, a lowering of RPV water level below the specified level indicates that operator actions have not been successful in restoring and maintaining RPV water level. The heat-up rate of the coolant will rise as the available water inventory is reduced. A continuing drop in water level will lead to core uncovery.

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

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

Reference(s):

1. Technical Requirements Table 3.3.5.1-1, Emergency Core Cooling System Instrumentation
2. Technical Requirements Table 3.3.5.2-1, RCIC Instrumentation
3. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Significant Loss of RPV inventory

EAL:

CA1.2 Alert

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

AND EITHER

- UNPLANNED rise in **any** Table C-1 sump or pool levels due to a loss of RPV inventory
- Visual observation of UNISOLABLE RCS leakage

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table C-1 Sumps/Pool
• Drywell equipment drain sump • Drywell floor drain sump • Pedestal floor drain sump • CTMT equipment drain sump • CTMT floor drain sump • Suppression Pool • RHR A, B, C, HPCS, LPCS, RCIC room sumps • Auxiliary Building floor drain sump

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

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

Basis:

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

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Rise in drywell equipment drain sump level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 1, 2). An Auxiliary Building sump level rise may also be indicative of RCS inventory losses external to the Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

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 RPV 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 RPV.

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

Reference(s):

1. SOP-0104 Floor and Equipment Drains System
2. SOP-0033 Drywell and Containment Leak Detection System
3. EOP-3 Secondary Containment and Radioactivity Release Control
4. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.1 Site Area Emergency

CONTAINMENT CLOSURE **not** established

AND

RPV water level < -143 in. (Level 1)

Mode Applicability:

4 – Cold Shutdown, 5 – Refueling

Definition(s):

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

Containment Closure is established when the containment requirements of OSP-0037 are met and at least one integral barrier to the release of radioactive material is provided, within the specified limits using STP-057-3804.

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

Basis:

The threshold RPV water level of -143 in. is the low-low-low ECCS actuation setpoint (Level 1). The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further lowering of RPV water level and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier (ref. 1, 2).

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

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV 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 RPV levels of EALs CS1.1 and CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

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

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

Reference(s):

1. Technical Requirements Table 3.3.5.1-1, Emergency Core Cooling System Instrumentation
2. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.2 Site Area Emergency
CONTAINMENT CLOSURE established
AND
RPV water level < -162 in. (TAF)

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

Containment Closure is established when the containment requirements of OSP-0037 are met and at least one integral barrier to the release of radioactive material is provided, within the specified limits using STP-057-3804.

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

Basis:

When RPV level drops to the top of active fuel (TAF) (an indicated RPV level of -162 in.), core uncovery starts to occur (ref. 1).

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

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV 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 RPV levels of EALs CS1.1 and CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

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

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

Reference(s):

1. EPSTG*0002 Appendix B EOP and SAP Bases
2. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.3 Site Area Emergency

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

AND

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

- UNPLANNED rise in **any** Table C-1 sump or pool levels of sufficient magnitude to indicate core uncovery
- Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery
- RMS-RE140 Refueling Floor Near North Entrance, RMS-RE141 Refueling Floor Near South Entrance or RMS-RE16 A/B Primary containment - PAM A/B reading > 9 R/hr

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table C-1 Sumps/Pool
• Drywell equipment drain sump • Drywell floor drain sump • Pedestal floor drain sump • CTMT equipment drain sump • CTMT floor drain sump • Suppression Pool • RHR A, B, C, HPCS, LPCS, RCIC room sumps • Auxiliary Building floor drain sump

Mode Applicability:

4 – Cold Shutdown, 5 – Refueling

Definition(s):

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

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

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

Basis:

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

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Rise in drywell equipment drain sump level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 1, 2). An Auxiliary Building sump level rise may also be indicative of RCS inventory losses external to the Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

In the Refueling Mode, as water level in the RPV lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in indications on area radiation monitors. The 9 R/hr value is selected for this EAL because it is 90% of the scale for RMS-RE140 and 141 (lower range monitors) and on scale for the higher range monitors. This value represents a reading that is higher than that likely to be attributable to normal refuel floor operations. These monitors are located in the Containment on the refuel floor.

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

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

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

The inability to monitor RPV 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 RPV.

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 AG1

Reference(s):

1. SOP-0104 Floor and Equipment Drains System
2. SOP-0033 Drywell and Containment Leak Detection System
3. EOP-3 Secondary Containment and Radioactivity Release Control
4. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CG1.1 General Emergency

RPV level < -162 in. (TAF) for ≥ 30 min. (Note 1)

AND

Any Containment Challenge indication, Table C-2

Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Table C-2 Containment Challenge Indications

- CONTAINMENT CLOSURE **not** established (Note 6)
- Drywell or containment hydrogen concentration > 4%
- UNPLANNED rise in containment pressure
- Exceeding one or more Secondary Containment Control MAX SAFE area radiation levels:

Area	DRMS Grid 2	Max. Safe Operating Value
RHR Equip Rm A	1213	9.5E+03 mR/hr
RHR Equip Rm B	1214	9.5E+03 mR/hr
RHR Equip Rm C	1215	9.5E+03 mR/hr

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

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

Containment Closure is established when the containment requirements of OSP-0037 are met and at least one integral barrier to the release of radioactive material is provided, within the specified limits using STP-057-3804.

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

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

Basis:

When RPV level drops below -162 in., core uncovery starts to occur (ref. 1).

Four conditions are associated with a challenge to Containment integrity:

- CONTAINMENT CLOSURE is not established.
- 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 hydrogen concentration has exceeded the minimum necessary to support a hydrogen burn (4%). The Igniter System is designed to prevent hydrogen accumulation by locally burning hydrogen in a controlled manner as soon as the hydrogen enters the containment atmosphere and reaches the igniters. For high rates of hydrogen production, ignition occurs at the lowest concentration that can support ignition. Following ignition, hydrogen is consumed through formation of diffusion flames where the gas enters the containment, thus controlling hydrogen concentration at approximately 4% (ref. 2).
- Any UNPLANNED rise in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of CONTAINMENT CLOSURE capability. UNPLANNED containment pressure rise indicates CONTAINMENT CLOSURE cannot be assured and the containment cannot be relied upon as a barrier to fission product release.
- Secondary Containment radiation monitors should provide indication of a larger release that may be indicative of a challenge to CONTAINMENT CLOSURE. The MAX SAFE radiation levels are indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in Table SC-2 of EOP-3, Secondary Containment

and Radioactivity Release Control that are in service under Cold Shutdown conditions (ref. 3).

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

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV 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.

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

Reference(s):

1. EOP-1 RPV Control
2. EPSTG*0002 Appendix B EOP and SAP Bases
3. EOP-3 Secondary Containment and Radioactivity Release Control
4. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CG1.2 General Emergency

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

AND

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

- UNPLANNED rise in **any** Table C-1 sump or pool levels of sufficient magnitude to indicate core uncovery
- Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery
- RMS-RE140 Refueling Floor Near North Entrance, RMS-RE141 Refueling Floor Near South Entrance or RMS-RE16 A/B Primary containment - PAM A/B reading > 9 R/hr

AND

Any Containment Challenge indication, Table C-2

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Table C-1 Sumps/Pool
• Drywell equipment drain sump • Drywell floor drain sump • Pedestal floor drain sump • CTMT equipment drain sump • CTMT floor drain sump • Suppression Pool • RHR A, B, C, HPCS, LPCS, RCIC room sumps • Auxiliary Building floor drain sump

Table C-2 Containment Challenge Indications

- CONTAINMENT CLOSURE **not** established (Note 6)
- Drywell or containment hydrogen concentration > 4%
- UNPLANNED rise in containment pressure
- Exceeding one or more Secondary Containment Control MAX SAFE area radiation levels:

Area	DRMS Grid 2	Max. Safe Operating Value
RHR Equip Rm A	1213	9.5E+03 mR/hr
RHR Equip Rm B	1214	9.5E+03 mR/hr
RHR Equip Rm C	1215	9.5E+03 mR/hr

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

Containment Closure is established when the Containment requirements of OSP-0037 (ref. 4.1.12) are met with the following exception: a functional barrier must exist at the time of the event (i.e., cannot rely on contingency methods to establish a functional barrier).

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

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

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

Basis:

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

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Rise in drywell equipment drain sump level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 1, 2). An Auxiliary Building sump level rise may also be indicative of RCS inventory losses external to the Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

In the Refueling Mode, as water level in the RPV lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in indications on area radiation monitors. The 9 R/hr value is selected for this EAL because it is 90% of the scale for RMS-RE140 and 141 (lower range monitors) and on scale for the higher range monitors. This value represents a reading that is higher than that likely to be attributable to normal refuel floor operations. These monitors are located in the Containment on the refuel floor.

Four conditions are associated with a challenge to Containment integrity:

- CONTAINMENT CLOSURE is not established.
- 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 hydrogen concentration has exceeded the minimum necessary to support a hydrogen

burn (4%). The Igniter System is designed to prevent hydrogen accumulation by locally burning hydrogen in a controlled manner as soon as the hydrogen enters the containment atmosphere and reaches the igniters. For high rates of hydrogen production, ignition occurs at the lowest concentration that can support ignition. Following ignition, hydrogen is consumed through formation of diffusion flames where the gas enters the containment, thus controlling hydrogen concentration at approximately 4% (ref. 3).

- Any UNPLANNED rise in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of CONTAINMENT CLOSURE capability. UNPLANNED containment pressure rise indicates CONTAINMENT CLOSURE cannot be assured and the containment cannot be relied upon as a barrier to fission product release.
- Secondary Containment radiation monitors should provide indication of a larger release that may be indicative of a challenge to CONTAINMENT CLOSURE. The MAX SAFE radiation levels are indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in Table SC-2 of EOP-3, Secondary Containment and Radioactivity Release Control that are in service under Cold Shutdown conditions (ref. 4).

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

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV 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 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 RPV 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 RPV.

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

Reference(s):

1. SOP-0104 Floor and Equipment Drains System
2. SOP-0033 Drywell and Containment Leak Detection System
3. EPSTG*0002 Appendix B EOP and SAP Bases
4. EOP-3 Secondary Containment and Radioactivity Release Control
5. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CU2.1 Unusual Event

AC power capability, Table C-3, to DIV I and DIV II 4.16 KV ENS buses 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table C-3 AC Power Sources
Offsite • 1RTX-XSR1C • 1RTX-XSR1D
Onsite • EGS-EG1A • EGS-EG1B

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, DEF - Defueled

Definition(s):

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

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

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

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

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

Basis:

The DIV III bus (1E22*S004) is not credited because it only supplies power to the HPCS pump and associated loads, not any long term decay heat removal systems. In particular, suppression pool cooling mechanisms would be essential subsequent to a station blackout.

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 greater 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 ENS bus. Some examples of this condition are presented below.

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

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

Reference(s):

1. USAR Section 8.1 Electric Power Introduction
2. EE-001AC Startup Electrical Distribution Chart
3. SOP-0046 4.16 KV System
4. AOP-0004 Loss of Offsite Power
5. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CA2.1 Alert

Loss of **all** offsite and **all** onsite AC power capability to DIV I and DIV II 4.16 KV ENS buses for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, DEF - Defueled

Definition(s):

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

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

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

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

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

Basis:

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.

Mitigative strategies using other power sources (Division III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as a Site Area Emergency because of the greater time available to restore an ENS 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.

This EAL is the cold condition equivalent of the hot condition EAL SS1.1.

Reference(s):

1. USAR Section 8.1 Electric Power Introduction
2. EE-001AC Startup Electrical Distribution Chart
3. SOP-0046 4.16 KV System
4. AOP-0004 Loss of Offsite Power
5. NEI 99-01 CA2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED rise in RCS temperature

EAL:

CU3.1	**Unusual Event**
UNPLANNED rise in RCS temperature to > 200°F due to loss of decay heat removal capability	

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

Containment Closure is established when the containment requirements of OSP-0037 are met and at least one integral barrier to the release of radioactive material is provided, within the specified limits using STP-057-3804.

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

Basis:

This IC addresses an UNPLANNED rise 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 EAL CA3.1.

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

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

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

Reference(s):

1. Technical Specification Table 1.1-1 Modes
2. STP-050-0700 RCS Pressure and Temperature Limits Verification
3. AOP-0051 Loss of Decay Heat Removal
4. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED rise in RCS temperature

EAL:

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5- Refueling

Definition(s):

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

Containment Closure is established when the containment requirements of OSP-0037 are met and at least one integral barrier to the release of radioactive material is provided, within the specified limits using STP-057-3804.

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

Basis:

This 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.

Reference(s):

1. EOP-1 RPV Control
2. Technical Specifications Table 1.1-1 Modes
3. STP-050-0700 RCS Pressure and Temperature Limits Verification
4. AOP-0051 Loss of Decay Heat Removal
5. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.1 Alert

UNPLANNED rise in RCS temperature to > 200°F for > Table C-4 duration (Note 1)

OR

UNPLANNED RPV pressure rise > 10 psig

Note 1: The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded. The Emergency Director is **not** allowed an additional 15 minutes to declare after the time limit is exceeded.

Table C-4 RCS Heat-up Duration Thresholds

RCS Status	CONTAINMENT CLOSURE Status	Heat-up Duration
Intact	N/A	60 min.*
Not intact	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:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

Containment Closure is established when the containment requirements of OSP-0037 are met and at least one integral barrier to the release of radioactive material is provided, within the specified limits using STP-057-3804.

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

Basis:

In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the RCS pressure rise criteria when the RCS is intact in Mode 4 or based on time to boil data when in Mode 5 or the RCS is not intact in Mode 4.

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

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

The RCS Heat-up Duration Thresholds table addresses a rise in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not intact. The 20-minute criterion was included to allow time for operator action to address the temperature rise.

The RCS Heat-up Duration Thresholds table also addresses a rise 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 rise without a substantial degradation in plant safety.

Finally, in the case where there is a rise in RCS temperature, the RCS is not intact 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 rise 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.

Reference(s):

1 Technical Specifications Table 1.1-1 Mods
2. STP-050-0700 RCS Pressure and Temperature Limits Verification
3. AOP-0051 Loss of Decay Heat Removal
4. NEI 99-01 CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – Loss of Vital DC Power

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

EAL:

CU4.1 Unusual Event

Indicated voltage is < 105 VDC on required Safety Related DIV I and DIV II 125 VDC buses for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

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

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

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

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

Basis

Safety Related DC buses ENB-SWG01A (DIV I) and ENB-SWG01B (DIV II) feed the Division I and Division II loads respectively. The Division I and Division II batteries each have 60 cells with a specific minimum voltage of 1.75 volts/cell. These cell voltages yield minimum design bus voltages of 105 VDC (ref. 1).

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

As used in this EAL, "required" means the vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if

Train A is out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an Unusual Event. A loss of Vital DC power to Train A would not warrant an emergency classification.

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

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

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

Reference(s):

1. Safety Related Battery Specification 244.521
2. USAR 8.3.2 DC Power Systems
3. NEI 99-01 CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.1 Unusual Event

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

OR

Loss of **all** Table C-5 State and local agency communication methods

OR

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

Table C-5 Communication Methods

System	Onsite	State/ Local	NRC
Plant radio system	X		
Plant Paging System	X		
Sound powered phones	X		
In-plant telephones	X		
Emergency Notification System (ENS)			X
Commercial Telephone System		X	X
Satellite Phones		X	X
State of Louisiana Radio		X	
State and Local Hotline radio		X	
INFORM Notification System		X	

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, DEF – Defueled

Definition(s):

None

Basis:

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

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

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

The second EAL condition addresses a total loss of the communications methods used to notify all State and local agencies of an emergency declaration. The State and local agencies referred to here are the Louisiana Department of Environmental Quality, Governor's Office of Homeland Security and Emergency Preparedness, five Local Parishes Office of Homeland Security and Emergency Preparedness and 24 hour notification points, Mississippi Emergency Management Agency and the Mississippi Highway Patrol.

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

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

Reference(s):

1. RBS Emergency Plan Section 13.3.6.1.5.4 Communications
2. RBS Emergency Plan Section 13.3.6.2.1 Site Communications
3. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting Safety Systems

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

EAL:

CA6.1 Alert

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

AND

Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode

AND EITHER:

- Event damage has caused indications of degraded performance to the second train of the SAFETY SYSTEM needed for the current operating mode
- Event damage has resulted in VISIBLE DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode

(Notes 9, 10)

Note 9: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then emergency classification is **not** warranted.

Note 10: If the hazardous event **only** resulted in VISIBLE DAMAGE, with **no** indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is **not** warranted.

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

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

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

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

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

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

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

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

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

VISIBLE DAMAGE - Damage to a SAFETY SYSTEM train that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected SAFETY SYSTEM train.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance;

commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

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

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

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

This EAL is the cold condition equivalent of hot condition EAL SA8.1.

Reference(s):

1. EP FAQ 2016-002
2. NEI 99-01 CA6

Category E – Independent Spent Fuel Storage Installation (ISFSI)

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

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

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

The RBS ISFSI is located wholly within the plant PROTECTED AREA. Therefore any security event related to the ISFSI are classified under Category H1 security event related EALs.

Category: ISFSI

Subcategory: Confinement Boundary

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1	**Unusual Event**

Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask (HI-STORM overpack) > **EITHER** of the following:

- 60 mRem/hr (ϒ + η) on the top of the overpack
- 600 mRem/hr (ϒ + η) on side of the overpack (excluding inlet and outlet ducts)

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY- The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the RBS ISFSI, Confinement Boundary is defined as the Holtec System Multi-Purpose Canister (MPC).

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI): A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

Basis:

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

The existence of "damage" is determined by radiological survey. The specified EAL threshold values correspond to 2 times the cask technical specification values. The technical specification (licensing bases document) multiple of "2 times", which is also used in Recognition Category A IC AU1, is used here to distinguish between non-emergency and emergency conditions (ref. 2). The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

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

Reference(s):

1. USAR 9.1.2.5 Holtec HI-STORM Dry Spent Fuel Storage System
2. RBS HI-STORM 100 SYSTEM Certificate of Compliance for Spent Fuel Storage Casks Amendment 5, Appendix A Technical Specifications for the HI-STORM 100 Cask System Section 5.7.4
3. NEI 99-01 E-HU1

Category F – Fission Product Barrier Degradation

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

EALs in this category represent threats to the defense in depth design concept that precludes the release of highly radioactive fission products to the environment. This concept relies on multiple physical barriers any one of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment. The primary fission product barriers are:

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

B. Reactor Coolant System Barrier (RCB): The RCS Barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping out to and including the isolation valves.

C. Containment Barrier (CNB): The Containment Barrier includes the drywell, the containment, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. Containment Barrier thresholds are used as criteria for escalation of the Emergency Classification Level (ECL) from an Alert to a Site Area Emergency or a General Emergency.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1. "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

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

Site Area Emergency:
Loss or potential loss of any two barriers

General Emergency:
Loss of any two barriers and loss or potential loss of third barrier

The logic used for emergency classification based on fission product barrier monitoring should reflect the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier.
- Unusual Event ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.
- For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC AG1 has been exceeded.
- The fission product barrier thresholds specified within a scheme reflect plant-specific RBS design and operating characteristics.
- As used in this category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside the primary containment, an interfacing system, or outside of the primary containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered to be RCS leakage.
- At the Site Area Emergency level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the 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: **Any** loss or **any** potential loss of either Fuel Clad or RCS

EAL:

FA1.1 **Alert**
Any loss or **any** potential loss of either Fuel Clad or RCS barrier (Table F-1)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds, bases and references.

At the Alert classification level, Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1

Reference(s):

1. NEI 99-01 FA1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss or potential loss of **any** two barriers

EAL:

FS1.1	**Site Area Emergency**
Loss or potential loss of **any** two barriers (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds, bases and references.

At the Site Area Emergency classification level, each barrier is weighted equally. A Site Area Emergency is therefore appropriate for any combination of the following conditions:

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

At the Site Area Emergency classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, the Emergency Director would have greater assurance that escalation to a General Emergency is less IMMINENT.

Reference(s):

1. NEI 99-01 FS1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss of **any** two barriers and loss or potential loss of third barrier

EAL:

FG1.1 General Emergency

Loss of **any** two barriers

AND

Loss or potential loss of the third barrier (Table F-1)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds, bases and references.

At the General Emergency classification level each barrier is weighted equally. A General Emergency is therefore appropriate for any combination of the following conditions:

- Loss of Fuel Clad, RCS and Containment Barriers
- Loss of Fuel Clad and RCS Barriers with potential loss of Containment Barrier
- Loss of RCS and Containment Barriers with potential loss of Fuel Clad Barrier
- Loss of Fuel Clad and Containment Barriers with potential loss of RCS Barrier

Reference(s):

1. NEI 99-01 FG1

Table F-1 Fission Product Barrier Threshold Matrix & Bases

Table F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Containment). The table is structured so that each of the three barriers occupies adjacent columns. Each fission product barrier column is further divided into two columns; one for Loss thresholds and one for Potential Loss thresholds.

The first column of the table (to the left of the Fuel Clad Barrier Loss column) lists the categories (types) of fission product barrier thresholds. The fission product barrier categories are:

A. RPV Water Level

B. RCS Leak Rate

C. Containment Conditions

D. Containment Radiation / RCS Activity

E. Containment Integrity or Bypass

F. 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 barrier column beginning with number one (ex., FCB1, FCB2…FCB6).

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 to category Loss followed by category Potential Loss beginning with Category A, then B,...,F.

Table F-1 Fission Product Barrier Threshold Matrix						
	Fuel Clad Barrier (FCB)		Reactor Coolant System Barrier (RCB)		Containment Barrier (CNB)	
Category	Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
A RPV Water Level	FCB1 SAP entry is required	FCB2 RPV water level **cannot** be restored and maintained > -162 in. (TAF) or **cannot** be determined	RCB1 RPV water level **cannot** be restored and maintained > -162 in. (TAF) or **cannot** be determined	None	None	CNB1 SAP entry is required
B RCS Leak Rate	None	None	RCB2 UNISOLABLE break in **any** of the following: • Main steam lines • RCIC steam Line • RWCU • Feedwater RCB3 Emergency Depressurization is required	RCB4 UNISOLABLE primary system leakage that results in exceeding **EITHER:** • One or more EOP-3 Max Normal area radiation operating value (Table F-2) • One or more Isolation Temperature alarms (Table F-2)	CNB2 UNISOLABLE primary system leakage that results in exceeding **EITHER:** • One or more EOP-3 Max Safe area radiation operating value that can be read in the Control Room (Table F-2) • One or more EOP-3 Max Safe area temperature operating value (Table F-2)	None
C CTMT Conditions	None	None	RCB5 Drywell pressure > 1.68 psid due to RCS leakage	None	CNB3 UNPLANNED rapid drop in containment pressure following containment pressure rise CNB4 Containment pressure response **not** consistent with LOCA conditions	CNB5 Containment pressure > 15 psig CNB6 Drywell or containment hydrogen concentration > 4% CNB7 Parameters **cannot** be restored and maintained within the safe zone of the HCTL curve (EOP Figure 2)
D CTMT Rad / RCS Activity	FCB3 Containment radiation (RMS-RE16) > 3,000 R/hr FCB4 Coolant activity > 300 μCi/gm dose equivalent I-131	None	RCB6 Drywell radiation (RMS-RE20) > 30 R/hr	None	None	CNB8 Containment radiation (RMS-RE16) > 12,000 R/hr
E CTMT Integrity or Bypass	None	None	None	None	CNB9 UNISOLABLE direct downstream pathway to the environment exists after Containment isolation signal CNB10 Intentional Containment venting per EOPs	None
F Emergency Director Judgment	FCB5 **Any** condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier	FCB6 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier	RCB7 **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	RCB8 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	CNB11 **Any** condition in the opinion of the Emergency Director that indicates loss of the Containment barrier	CNB12 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier

Barrier: Fuel Clad

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

FCB1
SAP entry is required

Definition(s):

None

Basis:

Emergency Operating Procedure (EOPs) specify entry to the Severe Accident Procedures (SAPs) when core cooling is severely challenged. These EOPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined (ref. 1, 2).

The EOP conditions requiring SAP entry represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

This threshold is also a Potential Loss of the Containment barrier (CNB1). Since SAP entry occurs after core uncovery has occurred a Loss of the RCS barrier exists (RCB1). SAP entry, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

The Loss threshold represents the EOP requirement for entry into the SAPs. This is identified in the BWROG EPGs/SAGs when adequate core cooling cannot be assured.

Reference(s):

1. EOP-1 RPV Control
2. EOP-4 RPV Flooding
3. EP FAQ 2015-004
4. NEI 99-01, RPV Water Level Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

FCB2

RPV water level **cannot** be restored and maintained > -162 in. (TAF) or **cannot** be determined

Definition(s):

None

Basis:

An RPV water level instrument reading of -162 in. indicates RPV level is at the top of active fuel (TAF) (ref. 1). When RPV level is at or above the TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV water level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops to TAF, the level is indicative of a challenge to core cooling and the Fuel Clad barrier.

When RPV water level cannot be determined, EOPs require entry to EOP-4, RPV Flooding. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained (ref. 2). When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-4 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in scram-failure events). If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the fuel clad barrier exists.

Note that EOP-1A, RPV Control, ATWS, may require intentionally lowering RPV water level to -162 in. and control level between the Minimum Steam Cooling RPV Water Level (MSCRWL) and the top of active fuel (ref. 3). Under these conditions, a high-power ATWS event exists and requires at least a Site Area Emergency classification in accordance with the System Malfunction - RPS Failure EALs.

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, EALs SA6.1 or SS6.1 will dictate the need for emergency classification.

This water level corresponds to the top of the active fuel and is used in the EOPs to indicate a challenge to core cooling.

The RPV water level threshold is the same as RCS barrier Loss threshold RCB1. Thus, this threshold indicates a Potential Loss of the Fuel Clad barrier and a Loss of the RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

This threshold is considered to be exceeded when, as specified in the site-specific EOPs, RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this Fuel Clad barrier Potential Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

The term “cannot be restored and maintained above” means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value cannot be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active fuel, but does not permit extended operation below the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

Since the loss of ability to determine if adequate core cooling is being provided presents a significant challenge to the fuel clad barrier, a potential loss of the fuel clad barrier is specified.

Reference(s):

1. EOP-1 RPV Control
2. EOP-4 RPV Flooding
3. EOP-1A RPV Control, ATWS
4 NEI 99-01 RPV Water Level Potential Loss 2.A

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

FCB3

Containment radiation (RMS-RE16) > 3,000 R/hr

Definition(s):

None

Basis:

The containment 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 significant fuel clad damage (ref. 1). Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold RCB6 since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the ECL to a Site Area Emergency.

There is no Fuel Clad barrier Potential Loss threshold associated with CTMT Radiation/RCS Activity.

Reference(s):

1. Calculation G13.18.9.4-045 Containment Doses for Emergency Action Levels (EALs)
2. NEI 99-01 Primary Containment Radiation Fuel Clad Loss 4.A

Barrier: Fuel Clad

Category: D. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

FCB4

Coolant activity > 300 µCi/gm dose equivalent I-131

Definition(s):

None

Basis:

This threshold indicates that RCS radioactivity concentration is greater than 300 µCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications.

There is no Fuel Clad barrier Potential Loss threshold associated with CTMT Radiation/RCS Activity.

Reference(s):

1. NEI 99-01 RCS Activity Fuel Clad Loss 1.A

Barrier: Fuel Clad

Category: D. CTMT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

FCB5
Any condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost.

Reference(s):

1. NEI 99-01 Emergency Director Judgment Fuel Clad Loss 6.A

Barrier: Fuel Clad

Category: F. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

FCB6
Any condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that are to be used by the Emergency Director 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.

Reference(s):

1. NEI 99-01 Emergency Director Judgment Fuel Clad Potential Loss 6.A

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

RCB1

RPV water level **cannot** be restored and maintained > -162 in. (TAF) or **cannot** be determined

Definition(s):

None

Basis:

An RPV water level instrument reading of -162 in. indicates level is at the top of active fuel (TAF) (ref. 1). TAF is significantly lower than the normal operating RPV level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Containment barriers, and initiation of all ECCS. If RPV water level cannot be maintained above TAF, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the lowering level trend. The cause of the loss of RPV inventory is therefore assumed to be a LOCA. By definition, a LOCA event is a Loss of the RCS barrier.

When RPV water level cannot be determined, EOPs require entry to EOP-4, RPV Flooding. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained (ref. 2). The instructions in EOP-4 specify emergency depressurization of the RPV, which is defined to be a Loss of the RCS barrier (RCS Loss RCB3).

Note that EOP-1A, RPV Control, ATWS, may require intentionally lowering RPV water level to -162 in. and control level between the Minimum Steam Cooling RPV Water Level (MSCRWL) and the top of active fuel (ref. 3). Under these conditions, a high-power ATWS event exists and requires at least a Site Area Emergency classification in accordance with the System Malfunction - RPS Failure EALs.

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs SA6 or SS6 will dictate the need for emergency classification.

This water level corresponds to the top of active fuel and is used in the EOPs to indicate challenge to core cooling.

The RPV water level threshold is the same as Fuel Clad barrier Potential Loss threshold FCB2. Thus, this threshold indicates a Loss of the RCS barrier and Potential Loss of the Fuel Clad barrier and that appropriately escalates the emergency classification level to a Site Area Emergency.

This threshold is considered to be exceeded when, as specified in the site-specific EOPs, RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this RCS barrier Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

The term, "cannot be restored and maintained above," means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value cannot be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active fuel, but does not permit extended operation beyond the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

There is no RCS barrier Potential Loss threshold associated with RPV Water Level.

Reference(s):

1. EOP-1 RPV Control
2. EOP-4 RPV Flooding
3. EOP-1A RPV Control, ATWS
4. NEI 99-01 RPV Water Level RCS Loss 2.A

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

RCB2

UNISOLABLE break in **any** of the following:

- Main steam line
- RCIC steam line
- RWCU
- Feedwater

Definition(s):

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

Basis:

Failure to isolate the leak (from the Control Room or locally), within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

The conditions of this threshold include required containment isolation failures allowing a flow path to the environment. A release pathway outside primary containment exists when flow is not prevented by downstream isolations. In the case of a failure of both isolation valves to close but in which no downstream flowpath exists, emergency declaration under this threshold would not be required. Similarly, if the emergency response requires the normal process flow of a system outside containment (e.g., EOP requirement to bypass MSIV low RPV water level interlocks and maintain the main condenser as a heat sink using main turbine bypass valves), the threshold is <u>not</u> met. The combination of these threshold conditions represent the loss of both the RCS and Containment (see Loss CNB9) barriers and justifies declaration of a Site Area Emergency (i.e., Loss or Potential Loss of any two barriers).

Even though RWCU and Feedwater systems do not contain steam, they are included in the list because an UNISOLABLE break could result in the high-pressure discharge of fluid that is flashed to steam from relatively large volume systems directly connected to the RCS.

Large high-energy lines that rupture outside primary containment can discharge significant amounts of inventory and jeopardize the pressure-retaining capability of the RCS until they are isolated. If it is determined that the ruptured line cannot be promptly isolated, remotely or locally, the RCS barrier Loss threshold is met.

Reference(s):

1. NEI 99-01 RCS Leak Rate RCS Loss 3.A

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

RCB3
Emergency Depressurization is required

Definition(s):

None

Basis:

Plant symptoms requiring Emergency RPV Depressurization per the EOPs are indicative of a loss of the RCS barrier (ref. 1, 2).

Emergency RPV Depressurization in accordance with the EOPs is indicative of a loss of the RCS barrier. If Emergency RPV Depressurization is performed, the plant operators are directed to open safety relief valves (SRVs). Even though the RCS is being vented into the suppression pool, a Loss of the RCS barrier exists due to the diminished effectiveness of the RCS to retain fission products within its boundary.

EOP-1 Emergency Depressurization allows terminating the depressurization if necessary to maintain RCIC as an injection source. This would require closing the SRVs. Even though the SRVs may be reclosed, this threshold is still met due to the requirement for an Emergency Depressurization having been met (ref. 2).

Reference(s):

1. EOP-1 RPV Control Emergency Depressurization
2. EP FAQ 2015-003
3. NEI 99-01 RCS Leak Rate RCS Loss 3.B

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

RCB4

UNISOLABLE primary system leakage that results in exceeding **EITHER**:

- One or more EOP-3 Max Normal area radiation operating value (Table F-2)
- One or more Isolation Temperature alarms (Table F-2)

Table F-2 Secondary Containment Operating Values		
Area Temperatures		
Parameter	Isolation Temperature	Max Safe
Main Steam Line Tunnel	173°F (P601-19A-A1/A3/B1/B3)	200°F
RHR Equipment Area 1 (A)	117°F (P601-20A-B4)	200°F
RHR Equipment Area 2 (B)	117°F (P601-20A-B4)	200°F
RCIC Equipment Area	182°F (P601-21A-B6)	200°F
RWCU Pump Room 1 (A) / 2 (B)	165°F (P680-1A-A2/B2)	200°F
Area Radiation Levels		
Parameter	Max Normal	Max Safe
HPCS Area (1212) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RHR Equipment Room A (1213) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RHR Equipment Room B (1214) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RHR Equipment Room C (1215) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
LPCS Equipment Room (1216) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
HPCS Penetration Area (1217) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
LPCS Penetration Area (1218) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RCIC Equipment Room (1219) Grid 2	1.20E+02 mR/hr	9.5E+03 mR/hr

Definition(s):

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

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

Basis:

Failure to isolate the leak (from the Control Room or locally), within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

The presence of elevated general area temperatures or radiation levels in the Secondary Containment may be indicative of UNISOLABLE primary system leakage outside the containment. The EOP-3 Max Normal and Isolation Temperature alarm setpoint values in Table F-2 define this RCS threshold because they are the maximum normal operating/ Technical Specification Isolation values and signify the onset of abnormal system operation. When parameters reach this level, equipment failure or mis-operation may be occurring. Elevated parameters may also adversely affect the ability to gain access to or operate equipment within the affected area. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-3, Secondary Containment and Radioactivity Release Control (ref. 1).

In general, multiple indications should be used to determine if a primary system is discharging outside Containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the Secondary Containment since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room FLOODING, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the Secondary Containment.

Potential loss of RCS based on primary system leakage outside the primary containment is determined from EOP temperature or radiation Max Normal Operating values in areas such as main steam line tunnel, RCIC, etc., which indicate a direct path from the RCS to areas outside primary containment.

A Max Normal Operating value is the highest value of the identified parameter expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly. An Isolation Temperature value is indicative of an UNISOLABLE leak when temperatures do not begin to recover as a result of the isolation actions following the alarm and represents a Technical Specification limiting value.

The indicators reaching the threshold barriers and confirmed to be caused by RCS leakage from a primary system warrant an Alert classification. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a reduction in the steam or water being discharged through an unisolated break in the system.

Reference(s):

1. EOP-3 Secondary Containment and Radioactivity Release Control
2. NEI 99-01 RCS Leak Rate RCS Potential Loss 3.A

Barrier: Reactor Coolant System

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

RCB5
Drywell pressure > 1.68 psid due to RCS leakage

Definition(s):

None

Basis:

The drywell high pressure scram setpoint is an entry condition to EOP-1, RPV Control. A high Containment pressure of greater than 0.3 psig is an entry condition to EOP-2, Primary Containment Control (ref. 1, 2). Normal containment pressure control functions (e.g., operation of drywell and containment cooling, vent using containment vessel purge, etc.) are specified in EOP-2 in advance of less desirable but more effective functions (e.g., Emergency Depressurization, etc.).

In the design basis, containment pressures above the drywell high pressure scram setpoint are assumed to be the result of a high-energy release into the containment for which normal pressure control systems are inadequate or incapable of reversing the rising pressure trend. Pressures of this magnitude, however, can be caused by non-LOCA events such as a loss of drywell cooling or inability to control containment vent/purge (ref. 3).

The threshold phrase "...due to RCS leakage" focuses the barrier failure on the RCS instead of the non-LOCA malfunctions that may adversely affect containment pressure. Drywell pressure greater than 1.68 psid with corollary indications (e.g., drywell temperature, indications of loss of RCS inventory) should therefore be considered a Loss of the RCS barrier. Loss of drywell cooling that results in pressure greater than 1.68 psid should not be considered an RCS barrier Loss.

The 1.68 psid value is the drywell high pressure setpoint which indicates a LOCA by automatically initiating the ECCS or equivalent makeup system.

There is no RCS barrier Potential Loss threshold associated with CTMT Conditions.

Reference(s):

1. EOP-1 RPV Control
2. EOP-2 Primary Containment Control
3. USAR Section 6.2.1 Containment Functional Design
4. NEI 99-01 Primary Containment Pressure RCS Loss 1.A

Barrier: Reactor Coolant System

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CTMT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

RCB6
Drywell radiation (RMS-RE20) > 30 R/hr

Definition(s):

N/A

Basis:

Under post-LOCA conditions coaxial cables used on the drywell post-accident monitors (RMS-RE20A/B) are susceptible to Thermally Induced Currents (TIC). These currents may cause the drywell PAMs to read falsely high (~469 R/hr) on a rapid temperature rise and read falsely low on a rapid temperature drop. When accident temperature conditions stabilize indicated radiation dose rates would be more accurate. The duration of the spurious signal would last approximately 15 minutes. During the period of false readings operators should rely on other indications of RCS leakage including a rise in drywell temperature and pressure (RCB5).

The drywell radiation monitor reading (38 R/hr rounded to 30 R/hr for readability) corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold FCB3 since it indicates a loss of the RCS Barrier only (ref. 1).

There is no RCS barrier Potential Loss threshold associated with CTMT Radiation/RCS Activity.

Reference(s):

1. Calculation G13.18.9.4-045 Containment Doses for Emergency Action Levels (EALs)
2. NEI 99-01 Primary Containment Radiation RCS Loss 4.A

Barrier: Reactor Coolant System

Category: D. CTMT Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

RCB7
Any condition in the opinion of the Emergency Director that indicates loss of the RCS barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is lost.

Reference(s):

- NEI 99-01 Emergency Director Judgment RCS Loss 6.A

Barrier: Reactor Coolant System

Category: F. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

RCB8

Any condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that may be used by the Emergency Director 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.

Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Potential Loss 6.A

Barrier: Containment

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

CNB1

SAP entry is required

Definition(s):

None

Basis:

EOPs specify entry to the SAPs when core cooling is severely challenged. These EOPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined (ref. 1, 2).

The EOP conditions requiring SAP entry represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

This threshold is also a Loss of the Fuel Clad barrier (Loss FCB1). Since SAP entry occurs after core uncovery has occurred a Loss of the RCS barrier exists (Loss RCB1). SAP entry, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

The Potential Loss threshold is identical to the Fuel Clad Loss RPV Water Level threshold FCB1. The Potential Loss requirement for entry into the SAPs indicates adequate core cooling cannot be assured and that core damage is possible. BWR EPGs/SAGs (RBS term SAPs) specify the conditions when the EPGs are exited and SAGs are entered. Entry into SAGs is a logical escalation in response to the inability to assure adequate core cooling.

PRA studies indicate that the condition of this Potential Loss threshold could be a core melt sequence which, if not corrected, could lead to RPV failure and greater potential for primary containment failure. In conjunction with the RPV water level Loss thresholds in the Fuel Clad and RCS barrier columns, this threshold results in the declaration of a General Emergency.

There is no Containment barrier Loss threshold associated with RPV Water Level.

Reference(s):

1. EOP-1 RPV Control
2. EOP-4 RPV Flooding
3. EP FAQ 2015-004
4. NEI 99-01 RPV Water Level PC Potential Loss 2.A

Barrier: Containment

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

CNB2

UNISOLABLE primary system leakage that results in exceeding **EITHER**:

- One or more EOP-3 Max Safe area radiation operating value (Table F-2)
- One or more EOP-3 Max Safe area temperature operating value (Table F-2)

Table F-2 Secondary Containment Operating Values		
Area Temperatures		
Parameter	Isolation Temperature	Max Safe
Main Steam Line Tunnel	173°F (P601-19A-A1/A3/B1/B3)	200°F
RHR Equipment Area 1 (A)	117°F (P601-20A-B4)	200°F
RHR Equipment Area 2 (B)	117°F (P601-20A-B4)	200°F
RCIC Equipment Area	182°F (P601-21A-B6)	200°F
RWCU Pump Room 1 (A) / 2 (B)	165°F (P680-1A-A2/B2)	200°F
Area Radiation Levels		
Parameter	Max Normal	Max Safe
HPCS Area (1212) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RHR Equipment Room A (1213) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RHR Equipment Room B (1214) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RHR Equipment Room C (1215) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
LPCS Equipment Room (1216) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
HPCS Penetration Area (1217) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
LPCS Penetration Area (1218) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RCIC Equipment Room (1219) Grid 2	1.20E+02 mR/hr	9.5E+03 mR/hr

Definition(s):

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

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

Basis:

Failure to isolate the leak (from the Control Room or locally), within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

The presence of elevated general area temperatures or radiation levels in the secondary containment may be indicative of UNISOLABLE primary system leakage outside the containment. The Max Safe conditions define this Containment barrier threshold because they are indicative of problems in the secondary containment that are spreading and pose a threat to achieving a safe plant shutdown. This threshold addresses problematic discharges outside containment that may not originate from a high-energy line break. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-3, Secondary Containment and Radioactivity Release Control (ref. 1).

In general, multiple indications should be used to determine if a primary system is discharging outside containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the secondary containment since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room FLOODING, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the secondary containment.

The Max Safe Operating Temperature and the Max Safe Operating Radiation Level are each the highest value of these parameters at which neither: (1) equipment necessary for the safe shutdown of the plant will fail, nor (2) personnel access necessary for the safe shutdown of the plant will be precluded. EOPs utilize these temperatures and radiation levels to establish conditions under which RPV depressurization is required.

There is no Containment barrier Potential Loss threshold associated with RCS Leak Rate.

Reference(s):

1. EOP-3 Secondary Containment and Radioactivity Release Control
2. NEI 99-01 RCS Leak Rate PC Loss 3.C

Barrier: Containment

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

CNB3

UNPLANNED rapid drop in containment pressure following containment pressure rise

Definition(s):

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

Basis:

Rapid UNPLANNED loss of containment pressure (i.e., not attributable to containment cooling or condensation effects) following an initial pressure rise indicates a loss of primary containment integrity.

This threshold relies on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

Reference(s):

1. NEI 99-01 Primary Containment Conditions PC Loss 1.A

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

CNB4
Containment pressure response **not** consistent with LOCA conditions

Definition(s):

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

Basis:

Primary containment pressure should rise as a result of mass and energy release into the primary containment from a LOCA. Thus, primary containment pressure not rising under these conditions indicates a loss of primary containment integrity.

These thresholds rely on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

Reference(s):

1. USAR Table 6.2-7, Results of Containment Response Analysis
2. USAR Table 6.2-1, Containment Design Parameters
3. NEI 99-01 Primary Containment Conditions PC Loss 1.B

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

CNB5
Containment pressure > 15 psig

Definition(s):

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

Basis:

When the containment pressure exceeds the maximum allowable value (15 psig) (ref. 1), containment venting may be required even if offsite radioactivity release rate limits will be exceeded (ref. 2). This pressure is based on the containment design pressure as identified in the accident analysis. If this threshold is exceeded, a challenge to the containment structure has occurred because assumptions used in the accident analysis are no longer VALID and an unanalyzed condition exists. This constitutes a Potential Loss of the Containment barrier even if a containment breach has not occurred.

The threshold pressure is the primary containment internal design pressure. Structural acceptance testing demonstrates the capability of the primary containment to resist pressures greater than the internal design pressure. A pressure of this magnitude is greater than those expected to result from any design basis accident and, thus, represent a Potential Loss of the Containment barrier.

Reference(s):

1. USAR Table 6.2-1, Containment Design Parameters
2. EOP-2 Primary Containment Control
3. NEI 99-01, Primary Containment Conditions PC Potential Loss 1.A

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

CNB6
Drywell or containment hydrogen concentration > 4%

Definition(s):

None

Basis:

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 hydrogen concentration has exceeded the minimum necessary to support a hydrogen burn (4%). The Igniter System is designed to prevent hydrogen accumulation by locally burning hydrogen in a controlled manner as soon as the hydrogen enters the containment atmosphere and reaches the igniters. For high rates of hydrogen production, ignition occurs at the lowest concentration that can support ignition. Following ignition, hydrogen is consumed through formation of diffusion flames where the gas enters the containment, thus controlling hydrogen concentration at approximately 4% (ref. 1).

If hydrogen concentration reaches or exceeds the lower flammability limit, as defined in plant EOPs, in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside the containment, loss of the Containment barrier could occur.

Reference(s):

1. EPSTG*0002 Appendix B EOP and SAP Bases
2. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.B

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

CNB7
Parameters **cannot** be restored and maintained within the safe zone of the HCTL curve (EOP Figure 2)

Definition(s):

None

Basis:

The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool temperature from which Emergency RPV Depressurization will not raise:

- Suppression chamber temperature above the maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized,

  OR

- Suppression chamber pressure above Primary Containment Pressure Limit, while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

The HCTL is a function of RPV pressure, suppression pool temperature and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant and therefore, the inability to maintain plant parameters below the limit constitutes a potential loss of containment.

The term "**cannot** be restored and maintained above" means the parameter value(s) is not able to be brought within the specified limit. The determination requires an evaluation of system performance and availability in relation to the parameter value and trend. A threshold prescribing declaration when a threshold value cannot be restored and maintained within a specified limit does not require immediate action simply because the current value is outside the limit, but does not permit extended operation outside the limit; the threshold must be considered reached as soon as it is apparent that operation within the limit cannot be attained.

Reference(s):

1. EOP-2 Primary Containment Control
2. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.C

Barrier: Containment

Category: D. CTMT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: C. CTMT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

CNB8
Containment radiation (RMS-RE16) > 12,000 R/hr

Definition(s):

None

Basis:

In order to reach this Containment barrier Potential Loss threshold, a loss of the RCS barrier (Loss RCB6) and a loss of the Fuel Clad barrier (Loss FCB3) have already occurred. This threshold, therefore, represents a General Emergency classification.

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that 20% of the fuel cladding has failed (ref. 1). 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 to a General Emergency.

There is no Containment barrier Loss threshold associated with CTMT Radiation/RCS Activity.

Reference(s):

1. Calculation G13.18.9.4-045 Containment Doses for Emergency Action Levels (EALs)
2. NEI 99-01 NEI 99-01 Primary Containment Radiation Potential Loss 4.A

Barrier: Containment

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

CNB9
UNISOLABLE direct downstream pathway to the environment exists after Containment isolation signal

Definition(s):

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

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

Basis:

Failure to isolate the leak (from the Control Room or locally), within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway directly to the environment. The concern is the UNISOLABLE open pathway to the environment. A failure of the ability to isolate any one line indicates a breach of containment integrity.

This threshold also applies to a containment bypass due to a HPCS or LPCS line break outside containment with injection check valve failure allowing an UNISOLABLE direct pathway for RCS release to the environment.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems or minor release pathways, such as instrument lines, not protected by the Primary Containment Isolation System (PCIS). Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include UNISOLABLE main steam line or RCIC steam line breaks, UNISOLABLE RWCU system breaks, and UNISOLABLE containment atmosphere vent paths. If the main condenser is available with an UNISOLABLE main steam line, there may be releases through the steam jet air ejectors and gland seal exhausters. These pathways are monitored, however, and do not meet the intent of an UNISOLABLE release path to the environment. These minor releases are assessed using the Category A, Abnormal Rad Levels / Rad Effluent, EALs.

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.

EOP-2 Primary Containment Control, may specify primary containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1). Under these conditions with a VALID containment isolation signal, the Containment barrier should be considered lost. Refer to CNB10.

Following the leakage of RCS mass into primary containment and a rise in primary containment pressure, there may be minor radiological releases associated with allowable primary containment leakage through various penetrations or system components. Minor releases may also occur if a primary containment isolation valve(s) fails to close but the primary containment atmosphere escapes to an enclosed system. These releases do not constitute a loss or potential loss of primary containment but should be evaluated using the Recognition Category A ICs.

There is no Containment barrier Potential Loss threshold associated with CTMT Integrity or Bypass.

Reference(s):

1. EOP-2 Primary Containment Control
2. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.A

Barrier: Containment

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

CNB10
Intentional Containment venting per EOPs

Definition(s):

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

Basis:

EOP-2, Primary Containment Control, may specify containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded. Under these conditions, with a VALID primary containment isolation signal, the threshold is met when the operator begins venting the containment in accordance with Enclosure 21, not when actions are taken to bypass interlocks prior to opening the vent valves (ref. 1).

Intentional venting of primary containment for primary containment pressure or combustible gas control to the secondary containment and/or the environment is a Loss of the Containment. Venting for primary containment pressure control when not in an accident situation (e.g., to control pressure below the drywell high pressure scram setpoint) does not meet the threshold condition.

There is no Containment barrier Potential Loss threshold associated with CTMT Integrity or Bypass.

Reference(s):

1. EOP-2 Primary Containment Control
2. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.B

Barrier: Containment

Category: E. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

CNB11

Any condition in the opinion of the Emergency Director that indicates loss of the Containment barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is lost.

Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

Barrier: Containment

Category: E. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

CNB12
Any condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment 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.

Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Potential Loss 6.A

Category H – Hazards and Other Conditions Affecting Plant Safety

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

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

1. Security

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

2. Seismic Event

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

3. Natural or Technological Hazard

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

4. Fire

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

5. Hazardous Gas

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

6. Control Room Evacuation

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

7. 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 which may warrant classification and subsequent implementation of the Emergency Plan, a provision for classification of emergencies based on operator/management experience and judgment is still necessary. The EALs of this category provide the Emergency Director the latitude to classify emergency conditions consistent with the established classification criteria based upon Emergency Director judgment.

Category: H – Hazards

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 RBS Security Shift Supervision

OR

Notification of a credible security threat directed at the site

OR

A validated notification from the NRC providing information of an aircraft threat

Mode Applicability:

All

Definition(s):

HOSTAGE - A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE ACTION - An act toward RBS 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 RBS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

OWNER CONTROLLED AREA (OCA) - For the purposes of classification, the Security Owner Controlled Area (SOCA) or the area between the SOCA Fence and the PROTECTED AREA Boundary.

PROJECTILE - An object directed toward a Nuclear Power Plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA - The area within the perimeter of the RBS security fence.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

SECURITY CONDITION - **Any** security event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A security condition does not involve a HOSTILE ACTION.

Basis:

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1 and HS1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program].*

The first threshold references the Security Shift Supervision because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.39 information.

The second threshold addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the Security Plan for RBS.

The third threshold addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with AOP-0063 Outside Threats (ref. 2).

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan for RBS (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

Reference(s):

1. RBS Security Plan
2. AOP-0063 Outside Threats
3. AOP-0054 Security Events
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

EAL:

HA1.1 Alert

A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by RBS Security Shift Supervision

OR

A validated notification from NRC of an aircraft attack threat within 30 min. of the site

Mode Applicability:

All

Definition(s):

HOSTAGE - A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE ACTION - An act toward RBS 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 RBS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

HOSTILE FORCE - One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

OWNER CONTROLLED AREA (OCA) - For the purposes of classification, the Security Owner Controlled Area (SOCA) or the area between the SOCA Fence and the PROTECTED AREA Boundary.

PROJECTILE - An object directed toward a Nuclear Power Plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA - The area within the perimeter of the RBS security fence.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the SECURITY 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.

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 EAL does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

The first threshold is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA.

The second threshold addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and OROs are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with AOP-0063 Outside Threats (ref. 2).

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 SECURITY OWNER CONTROLLED AREA was intentional (i.e., a HOSTILE ACTION). It is expected, although not certain, that notification by an appropriate Federal agency to the site would clarify this point. In this case, the appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. The emergency declaration, including one based on other ICs/EALs, should not be unduly delayed while awaiting notification by a Federal agency.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan for RBS (ref. 1).

Escalation of the emergency classification level would be via IC HS1.

Reference(s):

1. RBS Security Plan
2. AOP-0063 Outside Threats
3. AOP-0054 Security Events
4. NEI 99-01 HA1

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 RBS Security Shift Supervision

Mode Applicability:

All

Definition(s):

HOSTAGE - A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE ACTION - An act toward RBS 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 RBS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

HOSTILE FORCE - One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

OWNER CONTROLLED AREA (OCA) - For the purposes of classification, the Security Owner Controlled Area (SOCA) or the area between the SOCA Fence and the PROTECTED AREA Boundary.

PROJECTILE - An object directed toward a Nuclear Power Plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA - The area within the perimeter of the RBS security fence.

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. 1, 2, 3).

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program].*

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Site Area Emergency declaration will mobilize 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 EAL does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan for RBS (ref. 1).

Reference(s):

1. RBS Security Plan
2. AOP-0063 Outside Threats
3. AOP-0054 Security Events
4. NEI 99-01 HS1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE levels

EAL:

HU2.1 Unusual Event

Seismic event > OBE as indicated by **EITHER** of the following:

- Annunciator P680-02A-C06, SEISMIC EVENT HIGH
- Annunciator P680-02A-B06, SEISMIC EVENT HIGH/HIGH and amber lights illuminated on H13-P869 ERS-NBI101

Mode Applicability:

All

Definition(s):

None

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. The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the U. S. Geological Survey (USGS), check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

To avoid inappropriate emergency classification resulting from spurious actuation of the seismic instrumentation or felt motion not attributable to seismic activity, an offsite agency (USGS, National Earthquake Information Center (NEIC)) can confirm that an earthquake has occurred in the area of the plant. Such confirmation should not, however, preclude a timely emergency declaration based on receipt of the OBE alarm. If requested, provide the analyst with the following RBS coordinates: **30° 45' 26" north latitude, 91° 19' 54" west longitude** (ref. 3). Alternatively, near real-time seismic activity can be accessed via the NEIC website.

Reference(s):

1. ARP-680-02 P680-02 Alarm Response
2. AOP-0028 Seismic Event
3. USAR section 2.1.1.1 Specification of Location
4. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.1	**Unusual Event**
A tornado strike within the PROTECTED AREA	

Mode Applicability:

All

Definition(s):

PROTECTED AREA - The area within the perimeter of the RBS security fence.

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses a tornado striking (touching down) within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories A, F, S or C.

If damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under EAL CA6.1 or SA8.1.

A tornado striking (touching down) within the PROTECTED AREA warrants declaration of an Unusual Event regardless of the measured wind speed at the meteorological tower. A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

Reference(s):

1. AOP-0029 Severe Weather Operation
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.2 Unusual Event

Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component required by Technical Specifications for the current operating mode

Mode Applicability:

All

Definition(s):

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses FLOODING of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

Escalation of the emergency classification level would be based on ICs in Recognition Categories A, F, S or C.

Refer to EAL CA6.1 or SA8.1 for internal FLOODING affecting one or more SAFETY SYSTEM trains.

Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.3 Unusual Event

Movement of personnel within the PROTECTED AREA is IMPEDED due to an event external to the PROTECTED AREA involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

Mode Applicability:

All

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is **not** routinely employed).

PROTECTED AREA - The area within the perimeter of the RBS security fence.

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses a hazardous materials event originating at a location outside the PROTECTED AREA 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 A, F, S or C.

Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.4	**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

Definition(s):

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site FLOODING caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.

This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the FLOODING around the Cooper Station during the Midwest floods of 1993, or the FLOODING around Ft. Calhoun Station in 2011.

Escalation of the emergency classification level would be based on ICs in Recognition Categories A, F, S or C.

Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.1 Unusual Event

A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):

- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

The FIRE is located within **any** Table H-1 area

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table H-1 Fire Areas
• Reactor Building • Auxiliary Building • Fuel Building • Control Building • Standby Cooling Tower • Diesel Generator Building • Tunnels (B, D,E, F, G)

Mode Applicability:

All

Definition(s):

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do **not** constitute fires. Observation of flame is preferred but is **not** required if large quantities of smoke and heat are observed.

VALID - An indication, report, or condition, is considered to be valid when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

The 15 minute requirement begins with a credible notification that a FIRE is occurring, or receipt of multiple VALID fire detection system alarms or field validation of a single fire alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field.

Table H-1 Fire Areas are those areas that contain equipment necessary for safe operation and shutdown of the plant (ref. 1).

Reference(s):

1. AOP-0052 Fire Outside the Main Control Room in Areas Containing Safety Related Equipment
2. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.2 Unusual Event

Receipt of a single fire alarm (i.e., **no** other indications of a FIRE)

AND

The fire alarm is 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table H-1 Fire Areas
• Reactor Building • Auxiliary Building • Fuel Building • Control Building • Standby Cooling Tower • Diesel Generator Building • Tunnels (B, D,E, F, G)

Mode Applicability:

All

Definition(s):

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do **not** constitute fires. Observation of flame is preferred but is **not** required if large quantities of smoke and heat are observed.

VALID - An indication, report, or condition, is considered to be valid when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

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.

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."
>
> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.
>
> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in HU4.2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

The 30 minute requirement begins upon receipt of a single VALID fire detection system alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field. Actual field reports must be made within the 30 minute time limit or a classification must be made. If a fire is verified to be occurring by field report, classification shall be made based on EAL HU4.1, with the 15 minute requirement beginning with the verification of the fire by field report.

Table H-1 Fire Areas are those areas that contain equipment necessary for safe operation and shutdown of the plant (ref. 1).

Reference(s):

1. AOP-0052 Fire Outside the Main Control Room in Areas Containing Safety Related Equipment
2. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.3 Unusual Event

A FIRE within the 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

All

Definition(s):

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do **not** constitute fires. Observation of flame is preferred but is **not** required if large quantities of smoke and heat are observed.

PROTECTED AREA - The area within the perimeter of the RBS security fence.

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 EAL CA6.1 or SA8.1.

Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.4 Unusual Event

A FIRE within the PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

Mode Applicability:

All

Definition(s):

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do **not** constitute fires. Observation of flame is preferred but is **not** required if large quantities of smoke and heat are observed.

PROTECTED AREA - The area within the perimeter of the RBS security fence.

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 EAL CA6.1 or SA8.1.

Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gas

Initiating Condition: Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

HA5.1 Alert

Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 room or area

AND

Entry into the room or area is prohibited or IMPEDED (Note 5)

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then **no** emergency classification is warranted.

Table H-2 Safe Operation & Shutdown Rooms/Areas

Room/Area	Mode
Auxiliary Building 70' RHR B Pump Room	3
Auxiliary Building 80' RHR A Pump Room	3
Auxiliary Building 114' West	3
Control Building 95' Div 1 RSS Room	3

Mode Applicability:

3 – Hot Shutdown

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is **not** routinely employed).

Basis:

This IC addresses an event involving a release of a hazardous gas that precludes or IMPEDES access to equipment necessary to maintain normal plant operation, or required for a normal plant cooldown and shutdown. This condition represents an actual or potential substantial degradation of the level of safety of the plant.

An Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.

Evaluation of the IC and EAL does not require atmospheric sampling; it only requires the Emergency Director's judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly IMPEDE procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as IMPEDED if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

An emergency declaration is **not** warranted if any of the following conditions apply:

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 3.
- 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 access control measures are of a conservative or precautionary nature, and would not actually prevent or IMPEDE a required action.
- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

This EAL does not apply to firefighting activities that generate smoke and that automatically or manually activate a fire suppression system in an area.

Escalation of the emergency classification level would be via Recognition Category A, C or F ICs.

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).

EAL HA5.1 mode applicability has been limited to the mode limitations of Table H-2 (Mode 3 **only**).

Reference(s):

1. Attachment 2 Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases
2. NEI 99-01 HA5

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations

EAL:

HA6.1	**Alert**
An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels	

Mode Applicability:

All

Definition(s):

None

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Transfer of plant control begins when the last licensed operator leaves the Control Room.

Escalation of the emergency classification level would be via IC HS6.

Reference(s):

1. AOP-0031 Shutdown from Outside the Main Control Room
2. NEI 99-01 HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

Initiating Condition: Inability to control a key safety function from outside the Control Room

EAL:

HS6.1 Site Area Emergency

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels

AND

Control of **any** of the following key safety functions is **not** re-established within 15 min. (Note 1):

- Reactivity (Modes 1 and 2 **only**)
- RPV water level
- RCS heat removal

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refueling

Definition(s):

None

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on Emergency Director 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).

Transfer of plant control and the time period to establish control begins when the last licensed operator leaves the Control Room.

Escalation of the emergency classification level would be via IC FG1 or CG1

Reference(s):

1. AOP-0031 Shutdown from Outside the Main Control Room
2. EP FAQ 2015-014
3. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist that 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

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for an UNUSUAL EVENT.

Reference(s):

1. NEI 99-01 HU7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist that 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

Definition(s):

HOSTAGE - A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE ACTION - An act toward RBS 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 RBS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

OWNER CONTROLLED AREA (OCA) - For the purposes of classification, the Security Owner Controlled Area (SOCA) or the area between the SOCA Fence and the PROTECTED AREA Boundary.

PROJECTILE - An object directed toward a Nuclear Power Plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA - The area within the perimeter of the RBS security fence.

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.

Reference(s):

1. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist that 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

Definition(s):

HOSTILE ACTION - An act toward RBS 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 RBS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

OWNER CONTROLLED AREA (OCA) - For the purposes of classification, the Security Owner Controlled Area (SOCA) or the area between the SOCA Fence and the PROTECTED AREA Boundary.

PROJECTILE - An object directed toward a Nuclear Power Plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA - The area within the perimeter of the RBS security fence.

SITE BOUNDARY - For classification and dose projection purposes, the Site Boundary is the area defined as the exclusion area or exclusion zone in 10CFR100.3 (a) which is a boundary of approximately 3,000 feet (or 0.5748 mile) from the RBS reactor centerline.

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.

Reference(s):

1. NEI 99-01 HS7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist that 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

Definition(s):

HOSTAGE - A person(s) held as leverage against the station to ensure that demands will be met by the station.

HOSTILE ACTION - An act toward RBS 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 RBS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the OWNER CONTROLLED AREA).

IMMINENT - The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

OWNER CONTROLLED AREA (OCA) - For the purposes of classification, the Security Owner Controlled Area (SOCA) or the area between the SOCA Fence and the PROTECTED AREA Boundary.

PROJECTILE - An object directed toward a Nuclear Power Plant that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA - The area within the perimeter of the RBS security fence.

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.

Reference(s):

1. NEI 99-01 HG7

Category S – System Malfunction

EAL Group: Hot Conditions (RCS temperature > 200ºF); EALs in this category are applicable only in one or more hot operating modes.

Numerous system-related equipment failure events that warrant emergency classification have been identified in this category. They may pose actual or potential threats to plant safety.

The events of this category pertain to the following subcategories:

1. Loss of Emergency AC Power

Loss of emergency plant electrical power can compromise plant SAFETY SYSTEM operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite power sources for 4.16 KV ENS buses.

2. Loss of Vital DC Power

Loss of emergency plant electrical power can compromise plant SAFETY SYSTEM operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of power to or degraded voltage on the 125V DC vital buses.

3. Loss of Control Room Indications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant rise 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 pressure 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 scrams. In the plant licensing basis, postulated failures of the RPS to complete a reactor scram comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean any scram failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and containment integrity.

7. Loss of Communications

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

8. Hazardous Event Affecting Safety Systems

Various natural and technological events that result in degraded plant SAFETY SYSTEM performance or significant VISIBLE DAMAGE warrant emergency classification under this subcategory.

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite AC power capability to ENS buses for 15 minutes or longer

EAL:

SU1.1	**Unusual Event**
Loss of **all** offsite AC power capability, Table S-1, to DIV I and DIV II 4.16 KV ENS buses for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-1 AC Power Sources
Offsite
• 1RTX-XSR1C • 1RTX-XSR1D
Onsite
• EGS-EG1A • EGS-EG1B

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

The DIV III bus (1E22*S004) is not credited because it only supplies power to the HPCS pump and associated loads, not any long term decay heat removal systems. In particular, suppression pool cooling mechanisms would be essential subsequent to a station blackout.

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 ENS 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 ENS 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.

Reference(s):

1. USAR Section 8.1 Electric Power Introduction
2. EE-001AC Startup Electrical Distribution Chart
3. SOP-0046 4.16 KV System
4. AOP-0004 Loss of Offsite Power
5. NEI 99-01 SU1

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all but one** AC power source to ENS buses for 15 minutes or longer

EAL:

SA1.1 Alert

AC power capability, Table S-1, to DIV I and DIV II 4.16 KV ENS buses 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-1 AC Power Sources
Offsite
• 1RTX-XSR1C
• 1RTX-XSR1D
Onsite
• EGS-EG1A
• EGS-EG1B

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

The DIV III bus (1E22*S004) is not credited because it only supplies power to the HPCS pump and associated loads, not any long term decay heat removal systems. In particular, suppression pool cooling mechanisms would be essential subsequent to a station blackout.

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 ENS 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 ENS emergency power sources (e.g., onsite diesel generators) with a single train of ENS buses being back-fed from the unit main generator.
- A loss of ENS emergency power sources (e.g., onsite diesel generators) with a single train of ENS emergency buses being fed from an offsite power source.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

Escalation of the emergency classification level would be via IC SS1.

This EAL is the hot condition equivalent of the cold condition EAL CU2.1.

Reference(s):

1. USAR Section 8.1 Electric Power Introduction
2. EE-001AC Startup Electrical Distribution Chart
3. SOP-0046 4.16 KV System
4. AOP-0004 Loss of Offsite Power
5. NEI 99-01 SA1

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite and **all** onsite AC power to ENS buses for 15 minutes or longer

EAL:

SS1.1	**Site Area Emergency**
Loss of **all** offsite and **all** onsite AC power capability to DIV I and DIV II 4.16 KV ENS buses for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

This IC addresses a 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. Mitigative strategies using other power sources (Division III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout. 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 IC AG1, FG1 or SG1.

This EAL is the hot condition equivalent of the cold condition EAL CA2.1.

Reference(s):

1. USAR Section 8.1 Electric Power Introduction
2. EE-001AC Startup Electrical Distribution Chart
3. SOP-0046 4.16 KV System
4. AOP-0004 Loss of Offsite Power
5. NEI 99-01 SS1

Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to ENS buses

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power capability to DIV I and DIV II 4.16 KV ENS buses

AND EITHER:

- Restoration of at least one 4.16 KV ENS bus in < 4 hours is **not** likely (Note 1)
- RPV water level **cannot** be restored and maintained > -187 in.

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

Indication of continuing core cooling degradation is manifested by the inability to restore and maintain RPV water level above the Minimum Steam Cooling Reactor Water Level (-187 in.) (ref. 5). Core submergence is the most desirable means of core cooling, however when RPV level is below TAF, the uncovered portion of the core can be cooled by less reliable means (i.e., steam cooling or spray cooling).

This IC addresses a prolonged loss of all power sources to AC ENS 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. Mitigative strategies using other power sources (Division III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout. 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 ENS 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 a greater likelihood of challenges to multiple fission product barriers. 4 hours is the site-specific SBO coping analysis time (ref. 6).

The estimate for restoring at least one ENS 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.

Reference(s):

1. USAR Section 8.1 Electric Power Introduction
2. EE-001AC Startup Electrical Distribution Chart
3. SOP-0046 4.16 KV System
4. AOP-0004 Loss of Offsite Power
5. EOP-1 RPV Control
6. USAR Appendix 15C Station Blackout
7. NEI 99-01 SG1

Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** ENS AC and vital DC power sources for 15 minutes or longer

EAL:

SG1.2 General Emergency

Loss of **all** offsite and **all** onsite AC power capability to DIV I and DIV II 4.16 KV ENS buses for ≥ 15 min. (Note 1)

AND

Indicated voltage is < 105 VDC on Safety Related DIV I and DIV II 125 VDC buses for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

Safety Related DC buses ENB-SWG01A (DIV I) and ENB-SWG01B (DIV II) feed the Division I and Division II loads respectively. The Division I and Division II batteries each have 60 cells with a specific minimum voltage of 1.75 volts/cell. These cell voltages yield minimum design bus voltages of 105 VDC (ref. 5).

This IC addresses a concurrent and prolonged loss of both emergency ENS AC and Vital DC power. A loss of all emergency ENS 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. Mitigative strategies using other power sources (Division III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both emergency ENS AC and Vital DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

Reference(s):

1. USAR Section 8.1 Electric Power Introduction
2. EE-001AC Startup Electrical Distribution Chart
3. SOP-0046 4.16 KV System
4. AOP-0004 Loss of Offsite Power
5. Safety Related Battery Specification 244.521
6. USAR 8.3.2 DC Power Systems
7. NEI 99-01 SG8

Category: S – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

Initiating Condition: Loss of **all** vital DC power for 15 minutes or longer

EAL:

SS2.1	**Site Area Emergency**
Indicated voltage is < 105 VDC on Safety Related DIV I and DIV II 125 VDC buses for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

Safety Related DC buses ENB-SWG01A (DIV I) and ENB-SWG01B (DIV II) feed the Division I and Division II loads respectively. The Division I and Division II batteries each have 60 cells with a specific minimum voltage of 1.75 volts/cell. These cell voltages yield minimum design bus voltages of 105 VDC (ref. 1).

This IC addresses a loss of vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via IC AG1, FG1 or SG1.

This EAL is the hot condition equivalent of the cold condition EAL CU4.1.

Reference(s):

1. Safety Related Battery Specification 244.521
2. USAR 8.3.2 DC Power Systems
3. 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

EAL:

SU3.1 Unusual Event

An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-2 Safety System Parameters
• Reactor power
• RPV water level
• RPV pressure
• Containment pressure
• Suppression Pool water level
• Suppression Pool temperature

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

UNPLANNED - A parameter change or an event that is **not** 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

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, RPV water level 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 RPV water level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via EAL SA3.1.

Reference(s):

1. USAR 7.5 Safety-Related Display Instrumentation
2. NEI 99-01 SU2

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress

EAL:

SA3.1 Alert

An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

Any significant transient is in progress, Table S-3

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-2 Safety System Parameters
• Reactor power • RPV water level • RPV pressure • Containment pressure • Suppression Pool water level • Suppression Pool temperature

Table S-3 Significant Transients
• Reactor scram • Runback > 25% thermal reactor power • Electrical load rejection > 25% full electrical load • ECCS injection • Thermal power oscillations > 10%

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

UNPLANNED - A parameter change or an event that is **not** 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

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, RPV water level 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 RPV water level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via IC FS1 or AS1

Reference(s):

1. USAR 7.5 Safety-Related Display Instrumentation
2. NEI 99-01 SA2

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

SU4.1 Unusual Event

Offgas Pretreatment radiation monitor high alarm (P601-22A-F03, OFF GAS PRE-TREAT HIGH RADIATION)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

The Offgas Pretreatment monitors radioactivity in the Offgas system downstream of the Offgas condenser. The monitor detects the radiation level that is attributable to the fission gases produced in the reactor and transported with steam through the turbine to the condenser. The High alarm indicates that the radioactivity present at the recombiner effluent discharge is approaching the Technical Specification 3.7.4 limit. The nominal setpoint of 1.5 times the full power process background radiation level ensures that the activity will not exceed a value corresponding to the Technical Specification LCO 3.7.4 allowable release rate. (ref. 1)

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 IC FA1 or the Recognition Category A ICs.

Reference(s):

1. TRM section 3.3.7.8.2 Offgas System Radiation Monitoring Instrumentation
2. USAR 11.5 Process and Effluent Radiological Monitoring and Sampling Systems
3. Technical Specification 3.7.4 Main Condenser Offgas
4. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

SU4.2 Unusual Event

Coolant activity > 0.2 µCi/gm dose equivalent I-131 for > 48 hours

OR

Coolant activity > 4.0 µCi/gm dose equivalent I-131 instantaneous

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via IC FA1 or the Recognition Category A ICs.

Reference(s):

1. Technical Specification B 3.4.8, RCS Specific Activity bases
2. USAR Section 15.6.4 Steam System Piping Break Outside Containment
3. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1 Unusual Event

RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. (Note 1)

OR

RCS identified leakage > 25 gpm for ≥ 15 min. (Note 1)

OR

Leakage from the RCS to a location outside Containment > 25 gpm for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

UNISOLABLE - An open or breached system line that **cannot** be isolated, remotely or locally.

Basis:

Failure to isolate the leak (from the Control Room or locally) within 15 minutes, or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

Identified leakage is leakage into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a collecting sump; or leakage into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage.

Unidentified leakage is all leakage into the drywell that is not identified leakage (ref. 2, 3).

Pressure boundary leakage is leakage through a non-isolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall (ref. 2, 3).

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

The first and second EAL conditions are focused on a loss of mass from the RCS due to “unidentified leakage", "pressure boundary leakage" or "identified leakage” (as these leakage types are defined in the plant Technical Specifications). The third condition addresses an RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These conditions thus apply to leakage into the containment, or a location outside of containment.

The leak rate values for each condition were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). The first condition uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. 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 or F.

Reference(s):

1. USAR Section 5.2.5 Reactor Coolant Pressure Boundary and ECCS Leakage Detection System
2. Technical Specification Definitions Section 1.1
3. Technical Specification 3.4.5
2. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor

EAL:

SU6.1 Unusual Event

An automatic scram did **not** shut down the reactor as indicated by reactor power > 5% after **any** RPS setpoint is exceeded

AND

A subsequent automatic scram or manual scram action taken at the reactor control console (Mode Switch, Manual PBs, ARI) is successful in shutting down the reactor as indicated by reactor power ≤ 5% (APRM downscale)

Note 8: A manual scram 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, 2 - Startup

Definition(s):

IMMINENT - The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram 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.

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function. A reactor scram is automatically initiated by the RPS when certain continuously monitored parameters exceed predetermined setpoints (ref. 1).

A successful scram has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power to or below the APRM downscale setpoint of 5% (ref. 4).

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., mode switch, manual scram pushbuttons, or ARI initiation). Reactor shutdown achieved by use of alternate control rod insertion methods (i.e., EOP-1A Enclosure 26) does not constitute a successful manual scram (ref. 2).

Following any automatic RPS scram signal, operating procedures (e.g., EOP-1A) prescribe insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved. Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Unusual Event (ref. 3).

Taking the Mode Switch to Shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

For the purposes of this EAL, a successful automatic initiation of ARI that reduces reactor power to < 5% is not considered a successful automatic scram. If automatic initiation of ARI has occurred and caused reactor shutdown, the automatic RPS scram must have failed. ARI is a backup means of inserting control rods in the unlikely event that an automatic RPS scram signal exists but the reactor continues to generate significant power. However, a successful automatic or manual initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

In the event that the operator identifies a reactor scram is IMMINENT and initiates a successful manual reactor scram before the automatic scram setpoint is reached, no declaration is required. The successful manual scram of the reactor before it reaches its automatic scram setpoint or reactor scram signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. If manual reactor scram 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 scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

Following the failure on an automatic reactor scram, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram). 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 scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram using a different switch). Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles". Taking the Reactor Mode Switch to Shutdown is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via EAL SA6.1. 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.

Should a reactor scram signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal generated as a result of plant work causes a plant transient that results in a condition that should have included an automatic reactor scram and the RPS fails to automatically shutdown the reactor, then this IC and associated EALs are applicable, and should be evaluated.
- If the signal generated as a result of plant work does not cause a plant transient and the scram failure is determined through other means (e.g., assessment of test results), then this IC and associated EALs are not applicable and no classification is warranted.

Reference(s):

1. Technical Specification Table 3.3.1.1-1 Reactor Protection System Instrumentation
2. EOP-1A RPV Control, ATWS
3. EOP-1 RPV Control
4. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor

EAL:

SU6.2 Unusual Event

A manual scram did **not** shut down the reactor as indicated by reactor power > 5% after **any** manual scram action was initiated

AND

A subsequent automatic scram or manual scram action taken at the reactor control console (Mode Switch, Manual PBs, ARI) is successful in shutting down the reactor as indicated by reactor power ≤ 5% (APRM downscale) (Note 8)

Note 8: A manual scram 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, 2 - Startup

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram 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.

This EAL addresses a failure of a manually initiated scram in the absence of having exceeded an automatic RPS trip setpoint and a subsequent automatic or manual scram is successful in shutting down the reactor (reactor power ≤ 5%) (ref. 1).

A successful scram has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power to or below the APRM downscale setpoint of 5%.

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., Mode Switch, manual scram pushbuttons, or ARI initiation). Reactor shutdown achieved by use of alternate control rod insertion methods (i.e., EOP-1A Enclosure 26) does not constitute a successful manual scram (ref. 2).

Taking the Mode Switch to Shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

Successful automatic or manual initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

If both subsequent automatic and subsequent manual reactor scram 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 scram, the event escalates to an Alert under EAL SA6.1.

Following the failure on an automatic reactor scram, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram). 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 scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram) using a different switch. Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

Taking the Reactor Mode Switch to Shutdown is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

Should a reactor scram signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal generated as a result of plant work causes a plant transient that that results in a condition that should have included an automatic reactor scram and the RPS fails to automatically shutdown the reactor, then this IC and associated EALs are applicable, and should be evaluated.
- If the signal generated as a result of plant work does not cause a plant transient and the scram failure is determined through other means (e.g., assessment of test results), then this IC and associated EALs are not applicable and no classification is warranted.

Reference(s):

1. Technical Specification Table 3.3.1.1-1 Reactor Protection System Instrumentation
2. EOP-1A RPV Control, ATWS
3. EOP-1 RPV Control
4. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are **not** successful in shutting down the reactor

EAL:

SA6.1 Alert

An automatic or manual scram fails to shut down the reactor as indicated by reactor power > 5%

AND

Manual scram actions taken at the reactor control console (Mode Switch, Manual PBs, ARI) are **not** successful in shutting down the reactor as indicated by reactor power > 5% (Note 8)

Note 8: A manual scram 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, 2 - Startup

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

This EAL addresses any automatic or manual reactor scram signal that fails to shut down the reactor followed by a subsequent manual scram 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 (> 5%).

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., Mode Switch, manual scram pushbuttons, or ARI initiation). Reactor shutdown achieved by use of alternate control rod insertion methods (i.e., EOP-1A Enclosure 26) does not constitute a successful manual scram (ref. 2).

For the purposes of this EAL, a successful <u>automatic</u> initiation of ARI that reduces reactor power to or below 5% is <u>not</u> considered a successful automatic scram. If automatic actuation of ARI has occurred and caused reactor shutdown, the automatic RPS scram must have failed. ARI is a backup means of inserting control rods in the unlikely event that an automatic RPS scram signal exists but the reactor continues to generate significant power. However, a successful automatic initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). 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 (e.g., locally opening breakers). Actions taken at back panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

Taking the Reactor Mode Switch to Shutdown is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram) 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 shutdown the reactor is prolonged enough to cause a challenge to RPV water level 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.

Reference(s):

1. Technical Specification Table 3.3.1.1-1 Reactor Protection System Instrumentation
2. EOP-1A RPV Control, ATWS
3. EOP-1 RPV Control
4. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to RPV water level or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual scram 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:

RPV water level **cannot** be restored and maintained > -187 in.

OR

Heat Capacity Temperature Limit (HCTL) exceeded (EOP Figure 2)

Mode Applicability:

1 - Power Operation, 2 - Startup

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram 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.

This EAL addresses the following:

- Any automatic reactor scram signal followed by a manual scram that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the SAFETY SYSTEMS were designed (EAL SA6.1), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

Reactor shutdown achieved by use of control rod insertion methods in EOP-1A Enclosure 26 are also credited as a successful shutdown provided reactor power can be reduced to or below the APRM downscale trip setpoint before indications of an extreme challenge to either core cooling or heat removal exist. (ref. 1)

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.

Indication that core cooling is extremely challenged is manifested by inability to restore and maintain RPV water level above the Minimum Steam Cooling RPV Water Level (MSCRWL) (ref. 1). The MSCRWL is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500°F. This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core spray cooling could be a precursor of a core melt sequence (ref. 2).

The Heat Capacity Temperature Limit (HCTL, EOP Figure 2) is the highest suppression pool water temperature from which Emergency RPV Depressurization will not raise suppression pool temperature above the maximum design suppression pool temperature.

The HCTL is a function of RPV pressure and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant. This threshold is met when the final step of section SPT in EOP-2, Primary Containment Control, is reached (ref. 3). This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature.

In some instances, the emergency classification resulting from this 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 EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

Escalation of the emergency classification level would be via IC AG1 or FG1.

Reference(s):

1. EOP-1A, RPV Control, ATWS
2. EOP-4, RPV Flooding
3. EOP-2, Primary Containment Control
4. NEI 99-01 SS5

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

SU7.1 Unusual Event

Loss of **all** Table S-4 onsite communication methods

OR

Loss of **all** Table S-4 State and local agency communication methods

OR

Loss of **all** Table S-4 NRC communication methods

Table S-4 Communication Methods

System	Onsite	State/ Local	NRC
Plant radio system	X		
Plant Paging System	X		
Sound powered phones	X		
In-plant telephones	X		
Emergency Notification System (ENS)			X
Commercial Telephone System		X	X
Satellite Phones		X	X
State of Louisiana Radio		X	
State and Local Hotline radio		X	
INFORM Notification System		X	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to State and local agencies and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all State and local agencies of an emergency declaration. The State and local agencies referred to here are the Louisiana Department of Environmental Quality, Governor's Office of Homeland Security and Emergency Preparedness, Five Local Parishes Office of Homeland Security and Emergency Preparedness and 24 hour notification points, Mississippi Emergency Management Agency and the Mississippi Highway Patrol.

The third EAL condition addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

Reference(s):

1. RBS Emergency Plan Section 13.3.6.1.5.4 Communications
2. RBS Emergency Plan Section 13.3.6.2.1 Site Communications
3. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 8 – Hazardous Event Affecting Safety Systems

Initiating Condition: Hazardous event affecting SAFETY SYSTEMS needed for the current operating mode

EAL:

SA8.1 Alert

The occurrence of **any** Table S-5 hazardous event

AND

Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode

AND EITHER:

- Event damage has caused indications of degraded performance to the second train of the SAFETY SYSTEM needed for the current operating mode
- Event damage has resulted in VISIBLE DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode

(Notes 9, 10)

Note 9: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then emergency classification is **not** warranted.

Note 10: If the hazardous event **only** resulted in VISIBLE DAMAGE, with **no** indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is **not** warranted.

Table 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 Shift Manager

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should **not** automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do **not** constitute fires. Observation of flame is preferred but is **not** required if large quantities of smoke and heat are observed.

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

> Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:
>
> (1) The integrity of the reactor coolant pressure boundary;
>
> (2) The capability to shut down the reactor and maintain it in a safe shutdown condition;
>
> (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

VISIBLE DAMAGE - Damage to a SAFETY SYSTEM train that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected SAFETY SYSTEM train.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues.

Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make a determination of VISIBLE DAMAGE based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via IC FS1 or AS1.

This EAL is the hot condition equivalent of cold condition EAL CA6.1.

Reference(s):

1. EP FAQ 2016-00
2. NEI 99-01 SA9

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.

RBS Table A-3 and H-2 Bases

A review of station operating procedures identified the following mode dependent in-plant actions and associated areas that are required for normal plant operation, cooldown or shutdown:

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
GOP-0005 Power Operations			
Perform power maneuvering as directed by the OSM/CRS between 60 and 100% power using the guidance provided in the appropriate Reactivity Maneuvering Plan provided by Reactor Engineering.	MCR	1	
If possible, notify System Operator prior to changing generator load. • Adhere to MVAR vs. MW limits. • WHEN adjusting VARs on the Main Generator, THEN use VAR-1SPGN05 (H13-P680) only.	MCR	1	
Prior to entry into the Monitored Region of the Power/Flow map verify at least one PBDS Card is operable and begin STP-000-0001 monitoring of PBDS. (TS 3.3.1.3)	MCR	1	
Adjust pressure setpoint to minimize recirc pump "thrust reversals" as follows: • IF lowering power AND it is desired that pressure set be raised to minimize recirc pump "thrust reversals", THEN prior to lowering core flow to less than 70% rated core flow, raise reactor pressure. • IF raising power AND pressure set was raised to minimize recirc pump "thrust reversals", THEN when core flow is greater than 70% rated core flow, return the reactor pressure to its nominal value.	MCR	1	
Monitor Reactor Feed Pump vibration and flow. IF necessary to minimize vibration, THEN operate the reactor feed pump Minimum Flow valves per SOP-0009, Long Cycle Clean Up valve, or adjust reactor power.	MCR	1	
IF a reactor feed pump is anticipated to be shut down and Hydrogen injection will be left in service, THEN install the vent jumper for that pump per SOP-0009, Reactor Feedwater System.	MCR / TB 67'	1	Not required for plant shutdown or cooldown
Remove from service and/or restart Reactor Feed Pumps as necessary to maintain Reactor Water Level and reactor feed pump flow requirements to minimize vibration.	MCR / TB 67'	1	Not required for plant shutdown or cooldown
IF controlling Reactor Power with Reactor Recirculation Flow, THEN refer to SOP-0003.	MCR	1	
IF power is lowered below 75% AND a Reactor Feed Pump has been secured, THEN, BEFORE power ascension beyond 75% RTP and	MCR	1	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
AFTER start of the 3rd feedwater pump verify water chemistry acceptable.			
IF power is lowered below 75%, THEN when thermal power is above 75% RTP, verify both LEFM Check Plus are Operable.	MCR	1	
IF the LEFMs are functional per Technical Requirement 3.3.13 AND an LEFM alert is indicated on the ONE heat balance computer screen, THEN reduce reactor power to 3081 MWth or 99.6% rated thermal power in one hour to ensure thermal power limits are not exceeded.	MCR	1	
Observe the operation MSS-HVYCV4 and DTM-AOVSPDV4 for OSP-0102, Turbine Valve Testing. • As power is raised, check MSS-HVYCV4 open and DTM-AOVSPDV4 closed. • As power is lowered, check MSS-HVYCV4 closed and DTM-AOVSPDV4 open.	MCR	1	
When power is raised above 90%, Pressure Set may need to be adjusted as necessary to ensure that the 1st admission main turbine control valves, MSS-HVYCV1, 2, and 3 are full open.	MCR	1	
Monitor turbine vibration bearing temperature and differential expansion per the following: Turbine Temp & Expansion RCDR (TMI-NXR102) • Differential Expansion Rotor Long (point 11) between 0.31 inches and 0.69 inches. (Refer To ARP-870-54, G08, H08) • Rotor Expansion Rotor Long (point 12) between 0.455 inches and 1.545 inches. Turbine Vibration RCDR (TMI-NXR103) • Vibration (points 1 through 10) between 0 mils and 6 mils. (Refer To ARP-870-54, D08) Tamaris Computer (Display 69, 70) • Bearing oil temperatures (<setpoint, 180°F). • Bearing metal temperatures (<setpoint, 218.7°F). IF unusual indications are observed, THEN initiate hold in power change until those indications return to normal.	MCR	1	
WHEN maneuvering power, THEN adhere to the POWER/FLOW maps (avoid the restricted region) in AOP-0024, Thermal Hydraulic Stability Controls and Turbine-Generator loading rate per SOP-0080, Turbine Generator Operation.	MCR	1	
WHEN in two Recirculation Pump Operation at greater than or equal to 70% rated core flow, THEN maintain recirculation flow mismatch less than 5%.	MCR	1	
Observe the following limitations and precautions: • Do not exceed the Turbine Generator normal operating limits.	MCR	1	

GOP / SOP ACTIONS	LOCATION	MODE	NOTES

• Adjust the pressure setpoint to an indicated reactor pressure of between 1035 and 1055 psig for 100% steady state conditions.
• IF reactor power exceeds 3091 MWth, THEN take actions using recirculation flow and/or control rod insertion to lower power below 3091 MWth.
• IF the core thermal power average for a 2 hour period exceeds the Licensed Power Limit, THEN take timely action to ensure that thermal power is less than or equal to the Licensed Power Limit.
• IF reactor thermal power indication becomes unavailable for less than 15 minutes AND steady state operation is expected, THEN note current APRM readings AND verify thermal power does not exceed the noted value.
• IF reactor thermal power indication will be unavailable for more than 15 minutes, THEN perform the following:
  o Lower reactor power as indicated on the APRMs such that indicated thermal power does not exceed 100%. (The top of the normal noise band on the chart recorders should not be above 100%).
  o Reactor Engineering should be contacted for assistance in determining a manual heat balance per REP-0030, Reactor Heat Balance.
  o WHEN performing a manual heat balance AND it is determined that the LEFM signal is not operable, THEN lower reactor power so that the APRMs read less than 98.3% at the top of the normal noise band.
• Observe the following restrictions when operating near or above rated core flow as Bi-Stable flow conditions are possible:
  o IF step changes of 60 MWth (2%) or greater are seen in instantaneous CTP, THEN reduce Reactor power using Recirc flow until the step changes in instantaneous power are no longer observed.
  o IF step changes of up to 1.69 MLB/hr (2%) are seen in total core flow, THEN reduce Reactor power using Recirc flow until the step changes in instantaneous power are no longer observed.
  o Notify Reactor Engineering of any power/flow reductions required.
• IF any thermal limit exceeds 0.980, THEN notify Reactor Engineering to increase the frequency of monitoring (at least hourly) until a steady state condition is reached or thermal limits indicate less than 0.980.
• IF any thermal limit exceeds 0.990, THEN notify Reactor Engineering to perform one of the following:
  o Provide instructions for reducing the thermal limit to less than 0.990.
  o Provide a justification for operating with thermal limits			

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
greater than or equal to 0.990.			
IF the power change exceeded 15%, THEN perform the following: • Notify Chemistry of the power change to obtain a new Condensate System Oxygen injection flow rate. • Per Chemistry recommendations, adjust the Oxygen flow rate per SOP-0123, Hydrogen Water Chemistry H2 and O2 System. • IF power ramp rates exceed 15%/hr, THEN notify Chemistry per Technical Requirement.	MCR / TB 95'	1	Not required for plant shutdown or cooldown
IF power was lowered below 80%, THEN notify chemistry management when reactor power has been returned to 100%.	MCR	1	
As power is lowered, at approximately 50% power, **transfer Steam Seal Evaporator from Extraction Steam to Main Steam per SOP-0015**, Gland Seal Steam System and Exhaust System, if it has not occurred automatically.	MCR	1	
Transferring Steam Seal Evaporator from Extraction Steam to Main Steam (SOP-0015).			
As power is lowered, at approximately 50% power, if the Steam Seal Evaporator has not already transferred automatically from Extraction Steam to Main Steam, then throttle closed ESS-MOV112, STEAM SEAL EVAPORATOR using the control switch and the STOP pushbutton.	MCR	1	
IF the pressure controller is operating in automatic AND TME-MOVESFV2 is closed, THEN verify the following: • TME-PIEPR-35, SSE TUBE SIDE PRESSURE indicates less than or equal to 75 psig. • TME-PIEPR-36, SSE SHELL SIDE PRESSURE is stable and indicates less than or equal to 45 psig.	MCR	1	
WHEN ESS-MOV112, STEAM SEAL EVAPORATOR is full closed, THEN verify DTM-AOV118, EXTR STM TO SSE & RW RBLR opens.	MCR	1	
As power is raised, at approximately 65-75% power, after checking Annunciator P870-52-E03, 3rd PT EXTR ST AND MAIN STEAM DIFF PRESS LOW is clear, transfer Steam Seal Evaporator from Main Steam to Extraction Steam per SOP-0015, Gland Seal Steam System and Exhaust System.	MCR	1	
GOP-0002 Power Decrease/Plant Shutdown			
Notify System Operator prior to decreasing generator load.	MCR	1	
IF the Reference Leg Backfill System is not in service per SOP-0001, Nuclear Boiler Instrumentation (SYS #051), THEN have I&C stage equipment, acquire necessary technicians and obtain PMs to backfill	MCR	1	

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
reactor water level reference legs. (Approximately 12 hours may be needed to prepare for backfilling.) Actual backfilling performance may commence when Operations Shift Manager authorizes. (It is desired to have backfilling completed prior to reactor pressure reaching 450 psig to counter level indication notching possibilities.)			
Monitor turbine vibration bearing temperature and differential expansion per the following: Turbine Temp & Expansion RCDR (TMI-NXR102) • Differential Expansion Rotor Long (point 11) between 0.31 inches and 0.69 inches. (Refer To ARP-870-54, G08, H08) • Rotor Expansion Rotor Long (point 12) between 0.455 inches and 1.545 inches. Turbine Vibration RCDR (TMI-NXR103) • Vibration (points 1 through 10) between 0 mils and 6 mils. (Refer To ARP-870-54, D08) Tamaris Computer (Display 69, 70) • Bearing oil temperatures (<setpoint, 180°F). • Bearing metal temperatures (<setpoint, 218.7°F). IF unusual indications are observed, THEN initiate hold in power change until those indications return to normal.	MCR	1	
Lower reactor power per the Shutdown/ Emergency Power Reduction reactivity control plan. Contact the on-duty Reactor Engineer.	MCR	1	
Adjust pressure setpoint to minimize recirc pump "thrust reversals" as follows: • IF lowering power AND it is desired that pressure set be raised to minimize recirc pump "thrust reversals", THEN prior to lowering core flow to less than 70% rated core flow, raise reactor pressure. IF raising power AND pressure set was raised to minimize recirc pump "thrust reversals", THEN when core flow is greater than 70% rated core flow, return the reactor pressure to its nominal value.	MCR	1	
At approximately 90% to 80% power observe the following: • MSS-HVYCV4 closes • DTM-AOVSPDV4 opens	MCR	1	
IF MSRs are to be manually shutdown, THEN at approximately 90% power, start removing the MSRs from service per SOP-0010, MSR & FW Heaters Extraction Steam and Drains. Remove the MSRs at a rate so as to be completely off line by 760 MWe. Limit rate of change of LP Turbine inlet steam temperature to 125°F per hour. Monitor Points 6, 7, 8, 9 on TMI-NXR102. Maximum allowable temperature difference between LP Turbine inlets is 50°F. MSRs should be gradually valved	MCR / TB 123'	1	Not required for plant shutdown or cooldown

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
out in parallel at essentially the same temperature. IF MSRs are to remain in service with power maintained between 15% and 70%, THEN operate MSRs in accordance with SOP-0010.			
For power reductions of greater than 15%, notify Chemistry to determine whether the Condensate System oxygen injection is to be secured or flow reduced AND implement the recommendations per SOP-0123, Hydrogen Water Chemistry H2 and O2 System. IF power ramp rates exceed 15%/hr, THEN notify Chemistry per Technical Requirement 3.11.2.1.	MCR / TB 95'	1	Not required for plant shutdown or cooldown
IF a reactor feed pump is anticipated to be shut down AND Hydrogen Injection will be left in service AND a plant shutdown is NOT in progress, THEN install the vent jumper(s) for the pump(s) per SOP-0009, Reactor Feedwater System.	TB 67'	1	Not required for plant shutdown or cooldown
At approximately 70% power, (or with Engineering recommendations) **stop one reactor feedwater pump (leave two running) per SOP-0009, Reactor Feedwater System.**	MCR	1	
Reactor Feed Pump Shutdown (SOP-0009)			
IF securing a Reactor Feed Pump for downpower, THEN monitor the following parameters: • Reactor power should be limited to 85% with only two Reactor Feed Pumps in service. • Normal Feedwater Pump Motor current should be greater than 200 amps and limited to 311 amps. Refer to Precautions and Limitations 2.9 and 2.15. • FWREG position should be limited to less than or equal to 92% open to allow an adequate margin for valve modulation while maintaining reactor level. • Feed pump suction pressure should be maintained above low pressure alarm point of 280 psig.	MCR	1	
IF NOT already performed to reduce Reactor Feed Pump vibration levels, THEN perform the following for the Reactor Feed Pump being shutdown: • At H13-P680, place CNM-H/A68A(B)(C), RX FWP 1A(B)(C) MIN FLOW FLOW CONTROLLER to MANUAL for the Reactor Feed Pump to be secured. • Open slowly FWR-FV2A(B)(C), RX FWP 1A(B)(C) MIN FLOW VALVE using CNM-H/A68A(B)(C), RX FWP 1A(B)(C) MIN FLOW FLOW CONTROLLER while monitoring Reactor Water Level.	MCR	1	
IF desired to raise Reactor Water Level, THEN at H13-P680 adjust C33-R600, FW REG VALVES MASTER FLOW CONTROLLER tape set to desired Reactor Water Level within normal level control band.	MCR	1	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
IF the HWC System is in service AND the reactor feed pump is not being immediately shut down, THEN at P73-P500, place P73-AOV-F111A(B)(C), HYDROGEN ISOLATION TO FEEDWATER PUMP A(B)(C) in CLOSE.	MCR / TB 95'	1	Not required for plant shutdown or cooldown
IF the capability of meeting feed flow requirements with the remaining Feedwater Pumps is uncertain, THEN make a determination as follows: • Close FWS-MOV26A(B)(C), RX FWP 1A(B)(C) DISCH VLV for the pump being shutdown. • Verify the minimum flow valve for the pump being secured is open. • Monitor Feed Flow/Steam Flow mismatch and RPV Level to ensure remaining pump(s) can maintain level. • IF the remaining pump(s) cannot maintain RPV Level, THEN reopen the discharge valve FWS-MOV26A(B)(C), RX FWP 1A(B)(C) DISCH VLV and discontinue this procedure.	MCR	1	
IF the last Feedwater Pump is being removed from service, THEN open FWS-MOV109, FEED PUMP BYPASS.	MCR	1	
Stop FWS-P1A(B)(C), RX FWP P1A(B)(C).	MCR	1	
Verify CNM-H/A68A(B)(C), RX FWP 1A(B)(C) MIN FLOW FLOW CONTROLLER is in AUTO for the Reactor Feed Pump that was secured.	MCR	1	
IF Reactor Water Level was intentionally raised in Step 6.1.3, THEN adjust Reactor Water Level to desired level within normal level control band using C33-R600, FW REG VALVES MASTER FLOW CONTROLLER tape set.	MCR	1	
IF FWS-P1A(B)(C) is to remain in hot standby, THEN maintain seal temperatures as follows: Maintain seal water temperature dT less than or equal to 50F AND seal water outlet temperature less than or equal to 300F as follows: • FWS-P1A o Throttle CCS-V5003A, RFP FWL-P1A SEAL WATER HX-E4A CCS INLET VALVE, as required. o Throttle CCS-V5004A, RFP FWL-P1A SEAL WATER HX-E4B CCS INLET VALVE, as required. • FWS-P1B o Throttle CCS-V5003B, RFP FWL-P1B SEAL WATER HX-E4C CCS INLET VALVE, as required. o Throttle CCS-V5004B, RFP FWL-P1B SEAL WATER HX-E4D CCS INLET VALVE, as required. • FWS-P1C o Throttle CCS-V5003C, RFP FWL-P1C SEAL WATER HX-E4E CCS INLET VALVE, as required. o Throttle CCS-V5004C, RFP FWL-P1C SEAL WATER	TB 67'	1	Not required for plant shutdown or cooldown

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
HX-E4F CCS INLET VALVE, as required.			
On H13-P870, verify FWL-P5A(B)(C), GEAR INCR AUX OIL PMP 5A(B)(C) auto starts.	MCR	1	
Verify min flow valve closes 1 - 3 minutes after pump shutdown.	MCR	1	
Verify FWS-MOV26A(B)(C), RX FWP P1A (B)(C) DISCH VLV is closed.	MCR	1	
On H13-P870, WHEN the 23 minute time delay allowing for pump coast down has passed, THEN verify the following: • FWL-P1A(B)(C), RX FWP A(B)(C) MN OIL PMP 1A(B)(C) auto stops. • FWL-P5A(B)(C), RX FWP A(B)(C) GEAR INC AUX OIL PMP 5A(B)(C) auto stops. • FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) auto starts on low oil pressure. • IF FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) does not maintain pressure greater than 4 psi, THEN FWL-P3A(B)(C), RX FWP A(B)(C) AUX OIL PMP 3A(B)(C) auto starts on low oil pressure.	MCR	1	
On H13-P870, place FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) control switch in STOP, and verify FWL-P3A(B)(C), RX FWP A(B)(C) AUX OIL PMP 3A(B)(C) auto starts on low oil pressure.	MCR	1	
On H13-P870, place FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) control switch in AUTO.	MCR	1	
On H13-P870, place FWL-P3A(B)(C), RX FWP A(B)(C) AUX OIL PMP 3A(B)(C) control switch in STOP.	MCR	1	
On H13-P870, place FWL-P3A(B)(C), RX FWP A(B)(C) AUX OIL PMP 3A(B)(C) control switch in AUTO.	MCR	1	
Locally verify breaker relay trip flags are reset for Reactor Feed Pump stopped in Step 6.1.7.	NSW 98'	1	Not required for plant shutdown or cooldown
At approximately 70% power, (or with Engineering recommendations) **stop one condensate pump (leave two running) per SOP-0007, Condensate System.**	MCR	1	
Shutdown of CNM-P1A(B)(C) CONDENSATE PUMPS (SOP-0007)			
Request Aux Control Room remove unnecessary Condensate Filters from service per SOP-0124, Condensate Filtration System.	ACR	1	Not required for plant shutdown or cooldown
Request Aux Control Room remove unnecessary Condensate Demins from service per SOP-0093, Condensate Demineralizer System.	ACR	1	Not required for plant shutdown or

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
			cooldown
IF shutting down CNM-P1C, CNDS PUMP 1C, THEN secure Oxygen injection per SOP-0123, Hydrogen Water Chemistry H2 and O2 System.	MCR / TB 95'	1	Not required for plant shutdown or cooldown
IF securing the Condensate System, THEN perform the following: • Close all CNM-V3105 A, B, C, D, and E, CNM-FLT1 A, B, C, D, and E BACKWASH AIR SUPPLY valves. • IF desired to isolate and depressurize CNM-TK100, AIR RECEIVING TK, THEN perform the following: o CLOSE CNM-V3110, SVCE AIR ISOL VLV INLET SERV. AIR ISOL VLV. o Uncap and install hose on CNM-V3112, CNM-TK100 DRAIN ISOLATION VALVE. o Open CNM-V3112.	TB 123'	1	Not required for plant shutdown or cooldown
Depress the CLOSE pushbutton for CNM-MOV3A(B)(C), CNDS PUMP 1A(1B)(1C) DISCH.	MCR	1	
WHEN pump motor current lowers below 100 amps, THEN stop CNM-P1A(B)(C), CNDS PUMP 1A(1B)(1C).	MCR	1	
WHEN CNM-MOV3A(B)(C), CNDS PUMP 1A(1B)(1C) DISCH is full closed, THEN depress the STOP pushbutton.	MCR	1	
Verify associated CCS-MOV67A(B)(C), CNDS PMP 1A(1B)(1C) MOT CLR close for pump stopped.	MCR	1	
Verify associated CCS-MOV68A(B)(C), CNDS PMP 1A(1B)(1C) BRG CLR close for pump stopped.	MCR	1	
Locally verify breaker relay trip flags are reset for Condensate Pump stopped in Step 6.1.6.	NSW 98'	1	Not required for plant shutdown or cooldown
WHEN the Steam Jet Air Ejectors (SJAEs) and Gland Seal and Exhaust System are removed from service, THEN adjust CNM-H/A114 to 10% or to a setpoint determined by the CRS/OSM.	MCR	1	
As power is reduced, **remove FW Reg Valves from service per SOP-0009,** Reactor Feedwater System.	MCR	1	
Removing a FWREG Valve from Service (SOP-0009)			
Check feedwater flow is within the capability of the remaining FWREGs.	MCR	1	
Station an operator locally at the FWREG Valve to be removed from service.	TB 67'	1	Not required for plant shutdown or cooldown

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Establish communications between the local operator and the Main Control Room (MCR).	MCR / TB 67'	1	Not required for plant shutdown or cooldown
Place C33-R601A(R613)(R601B), FWREG VALVE A(B)(C) FLOW CONTROLLER in MANUAL.	MCR	1	
Throttle closed to 10% open the C33-LVF001A(B)(C), FWREG VALVE A(B)(C) to be removed from service while observing that reactor level is being maintained by the remaining FWREGs.	MCR	1	
IF level is not being maintained by the remaining FWREGs, THEN place the FWREG that was being removed from service back in service as follows: • Open C33-LVF001A(B)(C), FWREG VALVE A(B)(C) to the same position as the in service FWREGs. • Place C33-R601A(R613)(R601B), FWREG VALVE A(B)(C) FLOW CONTROLLER in AUTO.	MCR	1	
WHEN the FWREG is at 10% open, THEN close the following isolation valve for the FWREG valve that is being removed from service. • For C33-LVF001A close FWS-MOV27A, FWREG VLV 1A INLT Valve. • For C33-LVF001B close FWS-MOV27B, FWREG VLV 1B INLT Valve. • For C33-LVF001C close FWS-MOV27C, FWREG VLV 1C INLT Valve.	MCR	1	
Fully close the C33-LVF001A(B)(C), FWREG VALVE A(B)(C) that was removed from service.	MCR	1	
Record the temperature of the feedwater at the reactor feed pumps.	TB 67'	1	Not required for plant shutdown or cooldown
IF FWS-MOV27A, B, or C, FWREG VLV 1A(1B)(1C) INLT were closed with feedwater temperature at the reactor feed pumps greater than 200F, THEN refer to Section 5.7 for further stroking requirements.	MCR	1	
WHEN the FWREG Valve is at 0% open, THEN record demanded position in the MCR, position indication in the MCR, and local position indication.	MCR / TB 67'	1	Not required for plant shutdown or cooldown
Within one hour after reactor power is less than or equal to the high power setpoint, demonstrate RWL operability by performing STP-500-0704, Rod Withdrawal Limiter Functional Test (SR 3.3.2.1.2), if not performed within the previous 92 days.	MCR	1	STP-500-0704, Rod Withdrawal Limiter Functional Test is

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
			performed only in the MCR
Prior to entering the Monitored and/or the Restricted Regions of the Power to Flow map, verify the following indications on the PBDS, Period Based Detection System cards in APRM 'A' and 'B' cabinets: • NORMAL/BYPASS Toggle switch in the NORMAL position. • INOP STATUS LED indication is GREEN. (Depress the INOP STATUS Reset Pushbutton to reset a Red LED inop indication.) • Verify at least one PBDS Card is OPERABLE. • Begin STP-000-0001 monitoring of PBDS.	MCR	1	
Prior to entry into the Restricted Region of the Power to Flow Map, perform the following: • Verify FCBB is less than or equal to 1.0.(SR 3.2.4.1) • Place the APRM - FCTR, Flow Control Trip Reference cards to the setup trip setpoints by depressing the Normal/Setup pushbutton and verifying the normal/setup LED indication is yellow.	MCR	1	
At Approximately 50% power, **transfer Steam Seal Evaporator from Extraction Steam to Main Steam per SOP-0015, Gland Seal System And Exhaust System,** if it has not occurred automatically.	MCR	1	
Transferring Steam Seal Evaporator from Extraction Steam to Main Steam (SOP-0015)			
As power is lowered, at approximately 50% power, if the Steam Seal Evaporator has not already transferred automatically from Extraction Steam to Main Steam, then throttle closed ESS-MOV112, STEAM SEAL EVAPORATOR using the control switch and the STOP pushbutton.	MCR	1	
IF the pressure controller is operating in automatic AND TME-MOVESFV2 is closed, THEN verify the following: • TME-PIEPR-35, SSE TUBE SIDE PRESSURE indicates less than or equal to 75 psig. • TME-PIEPR-36, SSE SHELL SIDE PRESSURE is stable and indicates less than or equal to 45 psig.	MCR	1	
WHEN ESS-MOV112, STEAM SEAL EVAPORATOR is full closed, THEN verify DTM-AOV118, EXTR STM TO SSE & RW RBLR opens.	MCR	1	
At approximately 50% power, **shutdown all heater drain pumps per SOP-0010, MSR & FW Heaters Extraction Steam and Drains.**	MCR	1	
Removing the Heater Drain Pumps From Service (SOP-0010)			

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
IF it is desired to secure HDL-P1A(B), HTR DR PUMP 1A(B) for Heater String A, THEN at H13-P680, perform the following: • Depress the Close Pushbutton for HDL-MOV55A(B), HTR DR PMP 1A(B) DISCH. • Stop HDL-P1A(B), HTR DR PUMP 1A(B) for Heater String A. • Verify HDL-MOV55A(B), HTR DR PMP 1A(B) DISCH is closed.	MCR	1	
IF it is desired to secure HDL-P1C(D), HTR DR PUMP 1C(D) for Heater String B, THEN at H13-P680, perform the following: • Depress the Close Pushbutton HDL-MOV55C(D), HTR DR PMP 1C(D) DISCH. • Stop HDL-P1C(D), HTR DR PUMP 1C(D) for Heater String B. • Verify HDL-MOV55C(D), HTR DR PMP 1C(D) DISCH is closed.	MCR	1	
At approximately 50% power, perform the following per SOP-0006, Circulating Water, Cooling Tower and Vacuum Priming: • Shut down at least 1 circulating water pump. • Adjust the number of operating cooling tower fans to maintain vacuum and circulating water temperature.	MCR	1	
WHEN the recirculation flow control valves are at their minimum position, THEN continue reducing power by inserting control rods in their proper sequence.	MCR	1	
At about 40% power, **transfer both reactor recirculation pumps to SLOW speed per SOP-0003, Reactor Recirculation System.**	MCR	1	
Transferring from Fast Speed to Slow Speed (SOP-0003)			
Simultaneously depress B33-C001A and B RECIRC PUMP A and B MOTOR BREAKER 5A and 5B XFER TO LFMG pushbuttons.	MCR	1	
Observe the following: • Both B33-S001A LFMG MOT BRKR 1A and B33-S001B LFMG MOTBRKR 1B close. • Both B33-C001A RECIRC PUMP A MOTOR BREAKER 5A and B33-C001B RECIRC PUMP B MOTOR BREAKER 5B open. • WHEN B33-C001A and B, RECIRC PUMP A and B coast down to approximately 360 - 470 RPM, THEN B33-S001A and B LFMG A and B GEN BRKR 2A and 2B close and pump speeds stabilize near 450 RPM. • Both B33-K603 A and B, RECIRC LOOP A and B FLOW CONTROL MAN/AUTO stations transfer to MAN. • Open B33-HVY-F060A(B) to approximately 94% valve position using B33-K603A(B).	MCR	1	
Reduce to one reactor feed pump per SOP-0009, Reactor Feedwater System.	MCR	1	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Reactor Feed Pump Shutdown (SOP-0009)			
IF securing a Reactor Feed Pump for downpower, THEN monitor the following parameters: • Reactor power should be limited to 85% with only two Reactor Feed Pumps in service. • Normal Feedwater Pump Motor current should be greater than 200 amps and limited to 311 amps. Refer To Precautions and Limitations 2.9 and 2.15. • FWREG position should be limited to less than or equal to 92% open to allow an adequate margin for valve modulation while maintaining reactor level. • Feed pump suction pressure should be maintained above low pressure alarm point of 280 psig.	MCR	1	
IF NOT already performed to reduce Reactor Feed Pump vibration levels, THEN perform the following for the Reactor Feed Pump being shutdown: • At H13-P680, place CNM-H/A68A(B)(C), RX FWP 1A(B)(C) MIN FLOW FLOW CONTROLLER to MANUAL for the Reactor Feed Pump to be secured. • Open slowly FWR-FV2A(B)(C), RX FWP 1A(B)(C) MIN FLOW VALVE using CNM-H/A68A(B)(C), RX FWP 1A(B)(C) MIN FLOW FLOW CONTROLLER while monitoring Reactor Water Level.	MCR	1	
IF desired to raise Reactor Water Level, THEN at H13-P680 adjust C33-R600, FW REG VALVES MASTER FLOW CONTROLLER tape set to desired Reactor Water Level within normal level control band.	MCR	1	
IF the HWC System is in service AND the reactor feed pump is not being immediately shut down, THEN at P73-P500, place P73-AOV-F111A(B)(C), HYDROGEN ISOLATION TO FEEDWATER PUMP A(B)(C) in CLOSE.	MCR / TB 95'	1	Not required for plant shutdown or cooldown
IF the capability of meeting feed flow requirements with the remaining Feedwater Pumps is uncertain, THEN make a determination as follows: • Close FWS-MOV26A(B)(C), RX FWP 1A(B)(C) DISCH VLV for the pump being shutdown. • Verify the minimum flow valve for the pump being secured is open. • Monitor Feed Flow/Steam Flow mismatch and RPV Level to ensure remaining pump(s) can maintain level. • IF the remaining pump(s) cannot maintain RPV Level, THEN reopen the discharge valve FWS-MOV26A(B)(C), RX FWP 1A(B)(C) DISCH VLV and discontinue this procedure.	MCR	1	
IF the last Feedwater Pump is being removed from service, THEN open FWS-MOV109, FEED PUMP BYPASS.	MCR	1	
Stop FWS-P1A(B)(C), RX FWP P1A(B)(C).	MCR	1	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Verify CNM-H/A68A(B)(C), RX FWP 1A(B)(C) MIN FLOW FLOW CONTROLLER is in AUTO for the Reactor Feed Pump that was secured.	MCR	3	
IF Reactor Water Level was intentionally raised in Step 6.1.3, THEN adjust Reactor Water Level to desired level within normal level control band using C33-R600, FW REG VALVES MASTER FLOW CONTROLLER tape set.	MCR	3	
IF FWS-P1A(B)(C) is to remain in hot standby, THEN maintain seal temperatures as follows: Maintain seal water temperature dT less than or equal to 50F AND seal water outlet temperature less than or equal to 300F as follows: • FWS-P1A o Throttle CCS-V5003A, RFP FWL-P1A SEAL WATER HX-E4A CCS INLET VALVE, as required. o Throttle CCS-V5004A, RFP FWL-P1A SEAL WATER HX-E4B CCS INLET VALVE, as required. • FWS-P1B o Throttle CCS-V5003B, RFP FWL-P1B SEAL WATER HX-E4C CCS INLET VALVE, as required. o Throttle CCS-V5004B, RFP FWL-P1B SEAL WATER HX-E4D CCS INLET VALVE, as required. • FWS-P1C o Throttle CCS-V5003C, RFP FWL-P1C SEAL WATER HX-E4E CCS INLET VALVE, as required. o Throttle CCS-V5004C, RFP FWL-P1C SEAL WATER HX-E4F CCS INLET VALVE, as required.	TB 67'	1	Not required for plant shutdown or cooldown
On H13-P870, verify FWL-P5A(B)(C), GEAR INCR AUX OIL PMP 5A(B)(C) auto starts.	MCR	1	
Verify min flow valve closes 1 - 3 minutes after pump shutdown.	MCR	1	
Verify FWS-MOV26A(B)(C), RX FWP P1A (B)(C) DISCH VLV is closed.	MCR	1	
On H13-P870, WHEN the 23 minute time delay allowing for pump coast down has passed, THEN verify the following: • FWL-P1A(B)(C), RX FWP A(B)(C) MN OIL PMP 1A(B)(C) auto stops. • FWL-P5A(B)(C), RX FWP A(B)(C) GEAR INC AUX OIL PMP 5A(B)(C) auto stops. • FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) auto starts on low oil pressure. • IF FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) does not maintain pressure greater than 4 psi, THEN FWL-P3A(B)(C), RX FWP A(B)(C) AUX OIL PMP 3A(B)(C) auto starts on low oil pressure.	MCR	1	
On H13-P870, place FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) control switch in STOP, and verify FWL-P3A(B)(C), RX FWP	MCR	1	

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
A(B)(C) AUX OIL PMP 3A(B)(C) auto starts on low oil pressure.			
On H13-P870, place FWL-P2A(B)(C), RX FWP A(B)(C) AUX OIL PMP 2A(B)(C) control switch in AUTO.	MCR	1	
On H13-P870, place FWL-P3A(B)(C), RX FWP A(B)(C) AUX OIL PMP 3A(B)(C) control switch in STOP.	MCR	1	
On H13-P870, place FWL-P3A(B)(C), RX FWP A(B)(C) AUX OIL PMP 3A(B)(C) control switch in AUTO.	MCR	1	
Locally verify breaker relay trip flags are reset for Reactor Feed Pump stopped in Step 6.1.7.	NSW 98'	1	Not required for plant shutdown or cooldown
Decrease reactor power at the rate consistent with generator loading criteria (Attachment 1, MVAR VS MW LIMITS and SOP-0080, Turbine Generator Operation) using control rod insertion per applicable sequence. • Stop inserting control rods at the low power alarm point (as observed in RC & IS panel) and obtain instruction from the Operations Shift Manager regarding further power reductions/shutdown or continued operation at LPAP. • Decrease reactor power to the LPSP using control rod insertion per applicable sequence.	MCR	1	
At 300 MWe load, open the following steam drain valves: • DTM-AOV32A, 4TH PT HTR EXTR LINE DR • DTM-AOV32B, 4TH PT HTR EXTR LINE DR • DTM-AOV35A, 3RD PT HTR EXTR LINE DR • DTM-AOV35B, 3RD PT HTR EXTR LINE DR	MCR	1	
Open or verify open G33-MOVF101, RWCU BOTTOM HEAD DRAIN. • Verify drain temperature remains stable using Point #4 on B21R643 or ERIS computer point B33NA002.	MCR	1	
Manually stroke C33-LVF002, STARTUP FWREG VALVE through full travel to verify smooth operation **per SOP-0009, Reactor Feedwater System.**	MCR	1	
Manual Stroking of Start Up FWREG (SOP-0009)			
Close FWS-MOV105, S/U FW REG VLV ISOL.	MCR	1	
Station an operator locally to monitor valve position.	TB 67'	1	Not required for plant shutdown or cooldown

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Establish communications between the local operator and the Main Control Room (MCR).	MCR / TB 67'	1	Not required for plant shutdown or cooldown
WHEN the FWREG Valve is at 0% open, THEN record demanded position in the MCR, position indication in the MCR, and local position indication in Attachment 7, Calibration Check of FWREG Valves.	MCR / TB 67'	1	Not required for plant shutdown or cooldown
Use the OPEN and CLOSE Pushbuttons on C33-R602, START UP FWREG VALVE FLOW CONTROLLER to stroke open and then closed the Start Up FWREG.	MCR	1	
Check proper valve movement and smooth operation.	MCR / TB 67'	1	Not required for plant shutdown or cooldown
Check C33-LVF002, START UP FWREG VLV full closed.	MCR	1	
Open FWS-MOV105, S/U FW REG VLV ISOL.	MCR	1	
WHEN less than 30% power AND at the direction of the responsible Operations Management, THEN perform the following: • Transfer station loads to preferred source per SOP-0045, 13.8 KV System and SOP-0046, 4.16 KV System. • Verify MVARs are between + 50 and – 50. • At the SRM cabinets, place the Mode Selector Switches to the OPERATE position. • Prior to initiating a Rx Scram, verify the SRM & IRM Channel Functional Tests are current. IF Channel Functional Tests are not current, THEN refer to Tech Spec 3.3.1.1, 3.3.1.2 and TRM TR 3.3.2.1.	MCR	1	
Secure SPC per SOP-0140, Suppression Pool Cleanup and Alternate Decay Heat Removal.	MCR	1	Not required, but system will automaticall y isolate on a level 3 from a RX SCRAM
Contact the Auxiliary Control Room to verify that sufficient condensate demineralizers are in service to prevent physical damage to the demineralizers from high feedwater flow transients.	MCR / ACR	1/3	Not required for plant shutdown or cooldown

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Contact the Auxiliary Control Room to verify that sufficient condensate filtration filters are in service to prevent physical damage to the filters from high feedwater flow transients	MCR / ACR	1/3	Not required for plant shutdown or cooldown
Reduce the number of FWREG Valves in service to one per SOP-0009, Reactor Feedwater System.	MCR	1	Not required for plant shutdown or cooldown
Removing a FWREG Valve from Service (SOP-0009)			
Check feedwater flow is within the capability of the remaining FWREGs.	MCR	1	
Station an operator locally at the FWREG Valve to be removed from service.	TB 67'	1	Not required for plant shutdown or cooldown
Establish communications between the local operator and the Main Control Room (MCR).	MCR / TB 67'	1	Not required for plant shutdown or cooldown
Place C33-R601A(R613)(R601B), FWREG VALVE A(B)(C) FLOW CONTROLLER in MANUAL.	MCR	1	
Throttle closed to 10% open the C33-LVF001A(B)(C), FWREG VALVE A(B)(C) to be removed from service while observing that reactor level is being maintained by the remaining FWREGs.	MCR	1	
IF level is not being maintained by the remaining FWREGs, THEN place the FWREG that was being removed from service back in service as follows: • Open C33- LVF001A(B)(C), FWREG VALVE A(B)(C) to the same position as the in service FWREGs. • Place C33-R601A(R613)(R601B), FWREG VALVE A(B)(C) FLOW CONTROLLER in AUTO.	MCR	1	
WHEN the FWREG is at 10% open, THEN close the following isolation valve for the FWREG valve that is being removed from service • For C33-LVF001A close FWS-MOV27A, FWREG VLV 1A INLT Valve. • For C33-LVF001B close FWS-MOV27B, FWREG VLV 1B INLT Valve. • For C33-LVF001C close FWS-MOV27C, FWREG VLV 1C INLT Valve.	MCR	1	
Fully close the C33-LVF001A(B)(C), FWREG VALVE A(B)(C) that was removed from service.	MCR	1	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Record the temperature of the feedwater at the reactor feed pumps.	TB 67'	1	
IF FWS-MOV27A, B, or C, FWREG VLV 1A(1B)(1C) INLT were closed with feedwater temperature at the reactor feed pumps greater than 200F, THEN refer to Section 5.7 for further stroking requirements.	MCR	1	
WHEN the FWREG Valve is at 0% open, THEN record demanded position in the MCR, position indication in the MCR, and local position indication.	MCR / TB 67'	1	Not required for plant shutdown or cooldown
Line up RWCU reject to the main condenser **per SOP-0090, Reactor Feedwater System.**	MCR	1	Not required, but the preferred method to control level if shutdown long term.
RWCU blowdown operations (SOP-0090)			
Request chemistry sample to verify reactor water quality is within the specifications of Technical Requirement 3.4.13.	MCR	1	
Notify Radiation Protection prior to rejecting water to the Main Condenser or Radwaste.	MCR	1	
IF rejecting to the Main Condenser, THEN open G33 F046, RWCU DRAIN TO MN COND.	MCR	1	
IF rejecting during cold shutdown or refueling, THEN open G33 F031, RWCU DRAIN ORIFICE BYP.	MCR	1	
IF rejecting with the RWCU HXs isolated, THEN perform the following: • Open G33-F107, RWCU REGEN HX BYPASS. • Throttle open G33-PVF033, RWCU REJECT FLOW VALVE to establish reject flow as indicated on G33-R602, RWCU REJECT FLOW. • IF necessary to establish adequate reject flow, THEN close G33-F040, RWCU INBD RETURN VALVE.	MCR	1	
To establish the reject and maintain RWCU flowrate on G33 R609, RWCU INLET FLOW nearly constant, simultaneously throttle the following: • G33-PVF033, RWCU REJECT FLOW VALVE open using G33 R606, RWCU REJECT FLOW CONTROLLER • G33 F042, RWCU REGEN HX OUTLET closed	MCR	1	
Observe blowdown flow on G33 R602, RWCU REJECT FLOW.	MCR	1	
Monitor reactor water level while blowdown is in progress.	MCR	1	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
IF desired, THEN shutdown Reactor Recirculation HPU A(B) per SOP-0003 to prevent unnecessary Flow Control Valve movement.	MCR	1	
Initiate a Manual Scram per AOP-0001, Reactor Scram. • Verify the Hydrogen Water Chemistry (HWC) System shuts down on scram signal.	MCR	1/3	
WHEN it is desired to bypass the Feedwater Pump Level 8 Trip, THEN perform Attachment 5, Feedwater Pump Level 8 Trip Jumper Installation/Restoration Step 1.	MCR	3	Not required for plant shutdown or cooldown
WHEN it is desired to bypass the MSO Level 8 Trip, THEN perform Attachment 8, MSO Level 8 BYPASS Switch Step 1.	MCR	3	Not required for plant shutdown or cooldown
Monitor Bottom Head Drain Temperature on B21-R643 Point 4 or B33NA002 and take the following actions, as necessary, in a timely controlled manner to prevent an excessive temperature change. (STP-050-0700, RCS Pressure/Temperature Limits Verification). • Reset the Scram. • Reset any FCV runback per ARP-680-04.	MCR	3	
Within one hour after THERMAL POWER < 10% RTP in MODE 1, complete the following steps: • Verify/ensure that the RCIS data mode is selected to "CHAN 1 and CHAN 2". • Select and attempt to withdraw an out-of-sequence control rod. • Verify no rod motion occurs. • Verify Annunciator, P680-07A-C01, CONTROL ROD WITHDRAWAL BLOCK is actuated. • Verify WITHDRAWAL BLOCK Status Light is ON and not flashing. (SR 3.3.2.1.4)	MCR	3	
Place the APRM FCTR Cards to the Normal trip setpoints by depressing the Normal/Setup pushbutton and verifying the normal/setup LED indication is green	MCR	3	
After the Main Turbine is tripped, open the Feedwater Heater Vents per SOP-0010, MSR & FW Heaters Extraction Steam and Drains.	MCR	3	
Establish MSR Steam Blanketing per SOP-0010, MSR & FW Heaters Extraction Steam and Drains.	MCR	3	
Establishing Steam Blanketing (SOP-0010)			
IF Aux. Steam is available, THEN perform the following: • Throttle ASR-MOV104, MSR STM BLANKET SHUTOFF	MCR	3	

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
open until both red and green indication is received. • WHEN Aux steam to MSR steam blanketing line is warm as indicated on Computer point ASRTA01, THEN fully open ASR-MOV104. Verify the following are closed: • MSS-MOV111, MSR 1 STM SPLY SHUTOFF • MSS-MOV112, MSR 2 STM SPLY SHUTOFF • MSS-PVRSHLV1, MSR 1 HIGH LOAD VALVE • MSS-PVRSHLV2, MSR 2 HIGH LOAD VALVE • MSS-PVRSLLV1, MSR 1 LOW LOAD VALVE • MSS-PVRSLLV2, MSR 2 LOW LOAD VALVE • DSR-MOV107, SCAV STM TO 1ST PT HTR A • DSR-MOV109, SCAV STM TO 1ST PT HTR B • DSR-MOV108, SCAV STM TO COND A • 10) DSR-MOV110, SCAV STM TO COND B • DTM-MOV54A, MSL TO MSR 1 COND DR • DTM-MOV54B, MSL TO MSR 2 COND DR IF Aux Steam is available, THEN open the following: • ASR-MOVBSFV1, MSR 1 STM BLANKET SPLY • ASR-MOVBSFV2, MSR 2 STM BLANKET SPLY WHEN several minutes have elapsed after opening ASR-MOVBSFV1 and ASR-MOVBSFV2, THEN place the following control switches to CLOSE: • MSS-MOV111, MSR 1 STM SPLY SHUTOFF • MSS-MOV112, MSR 2 STM SPLY SHUTOFF			
IF notching is observed during the depressurization and magnitude is less than six inches, THEN: • Make all possible attempts to maintain reactor pressure. • Have I&C backfill the reference leg in which notching was observed, even if reference leg was overfilled prior to this event. IF notching is observed during the depressurization and magnitude is greater than six inches, THEN declare the trip channels associated with that signal inoperable and comply with Technical Specification requirements.	MCR	3	Not required for plant shutdown or cooldown
Reduce the number of running condensate pumps to one per SOP-0007, Condensate System.	MCR	3	
Shutdown of CNM-P1A(B)(C) CONDENSATE PUMPS (SOP-0007)			
Request Aux Control Room remove unnecessary Condensate Filters from service per SOP-0124, Condensate Filtration System.	ACR	1	Not required for plant shutdown or cooldown

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Request Aux Control Room remove unnecessary Condensate Demins from service per SOP-0093, Condensate Demineralizer System.	ACR	1	Not required for plant shutdown or cooldown
IF shutting down CNM-P1C, CNDS PUMP 1C, THEN secure Oxygen injection per SOP-0123, Hydrogen Water Chemistry H2 and O2 System.	MCR / TB 95'	1	Not required for plant shutdown or cooldown
IF securing the Condensate System, THEN perform the following: • Close all CNM-V3105 A, B, C, D, and E, CNM-FLT1 A, B, C, D, and E BACKWASH AIR SUPPLY valves. • IF desired to isolate and depressurize CNM-TK100, AIR RECEIVING TK, THEN perform the following: o CLOSE CNM-V3110, SVCE AIR ISOL VLV INLET SERV. AIR ISOL VLV. o Uncap and install hose on CNM-V3112, CNM-TK100 DRAIN ISOLATION VALVE. o Open CNM-V3112.	TB 123'	1	Not required for plant shutdown or cooldown
Depress the CLOSE pushbutton for CNM-MOV3A(B)(C), CNDS PUMP 1A(1B)(1C) DISCH.	MCR	1	
WHEN pump motor current lowers below 100 amps, THEN stop CNM-P1A(B)(C), CNDS PUMP 1A(1B)(1C).	MCR	1	
WHEN CNM-MOV3A(B)(C), CNDS PUMP 1A(1B)(1C) DISCH is full closed, THEN depress the STOP pushbutton.	MCR	1	
Verify associated CCS-MOV67A(B)(C), CNDS PMP 1A(1B)(1C) MOT CLR close for pump stopped.	MCR	1	
Verify associated CCS-MOV68A(B)(C), CNDS PMP 1A(1B)(1C) BRG CLR close for pump stopped.	MCR	1	
Locally verify breaker relay trip flags are reset for Condensate Pump stopped in Step 6.1.6	NSW 98'		Not required for plant shutdown or cooldown
WHEN the Steam Jet Air Ejectors (SJAEs) and Gland Seal and Exhaust System are removed from service, THEN adjust CNM-H/A114 to 10% or to a setpoint determined by the CRS/OSM.	MCR	1	
WHEN less than or equal to 145 MWT, THEN perform the following: • Secure SJAE per SOP-0092, Offgas System.	MCR	1	
• Start a mechanical vacuum pump per SOP-0025, Condenser Air Removal System.	TB 123' & 95'		Not required for plant shutdown or

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
			cooldown
WHEN all rods have been fully inserted, THEN initiate RPV cooldown at less than or equal to 100°F/hr by one of the following methods: • Initiate an automatic cooldown of the RPV from HMI Screen 5532, Pressure Control by performing the following: o Depress AUTO RATE in the Cooldown/Heatup Rate section and verify the pushbutton turns cornsilk and disabled. o Entered the desired cooldown rate and depress ENTER. o Enter the desired Target Pressure in the Throttle Pressure Control section and verify the Target pressure reflects the value entered. o Depress the GO pushbutton and verify: ▪ Pressure regulator setpoint is changing automatically. ▪ Bypass valves modulate to control pressure. ▪ RPV temperature lowers in accordance with the selected cooldown rate. • Initiate a cooldown by slowing and periodically reducing turbine pressure regulator setpoint from HMI Screen 5532, Pressure Control as follows: o Enter the desired Target pressure into the HMI display and depress ENTER. o Depress the GO pushbutton and verify Press Set changes to reflect the Target value. • ON HMI Screed 5537, BPV Jack, control the Bypass valves as follows to establish the required steam flow to the main condenser for cooldown: o Enter the BPV Jack Rate and depress ENTER. o Enter the BPV Target and depress ENTER. o Verify the BPVs open to the desired Target value to establish steam flow to the main condenser.	MCR	3	
Prior to reaching 600 psig, initiate monitoring of ERIS Narrow Range level for "notching" of one or more level indications (IF ERIS trending is not available, THEN contact I&C to arrange for alternate Narrow Range trending).	MCR	3	
Down range IRMs to maintain indication between downscale alarm and upscale alarm.	MCR	3	
Insert SRMs to maintain SRM counts between 1 x 103 and 1 x 105 cps. Fully insert SRMs before the IRMs are on range 3.	MCR	3	
Verify overlap between SRM and IRM. (All SRMs reading < 1x105 cps	MCR	3	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
prior to IRMs reading < 5 on range 1.)			
WHEN all control rods are fully inserted, THEN perform one of the following: • NORMAL SHUTDOWN ○ Place the REACTOR SYSTEM MODE SWITCH to SHUTDOWN. ○ WHEN at least 10 seconds have elapsed, THEN reset the Reactor Scram. • SOFT SHUTDOWN ○ Bypass the REACTOR MODE SWITCH POSITION SCRAM per SOP-0079, Reactor Protection System Attachment 7. ○ Place the REACTOR SYSTEM MODE SWITCH to SHUTDOWN. ○ WHEN at least 10 seconds have elapsed, THEN restore the REACTOR MODE SWITCH POSITION SCRAM per SOP-0079, Reactor Protection System Attachment 7.	MCR	3	
WHEN the reactor is shutdown AND at the direction of the Operations Shift Manager, THEN perform a drywell inspection per Attachment 3, Drywell Inspection Checklist.	DW 141'/131'/118'/107'/95'/82'	3	Not required for plant shutdown or cooldown
Maintain hot shutdown condition with RPV pressure between 250 psig and 1055 psig.	MCR	3	
IF RHR Pump warmup and flushing is required, THEN perform warmup/flushing per SOP-0031, Residual Heat Removal.	MCR	3	
Shutdown Cooling Flush, Warmup, and Startup (SOP-0031)			
IF RPV pressure is less than 135 psig, THEN have Electrical Maintenance implement the PM Task to re-land the thermal overload/loss of power annunciator leads for E12-F009, RHR SHUTDOWN COOLING INBD ISOL VALVE.	MCR	3	
Verify the following breakers are ON: • EHS-MCC2E BKR 5C, C002A DISCH MIN FLOW VALVE • EHS-MCC2F BKR 7B, C002B DISCH MIN FLOW	MCR	3	
Shutdown Cooling Flush • Request Chemistry to verify Suppression Pool is within best practice limits of CSP-0006, Chemistry Surveillance and Scheduling System. • IF suppression pool conductivity is NOT within best practice limits of CSP-0006, THEN perform a complete flush. • IF desired and suppression pool conductivity is within best practice limits of CSP-0006, THEN perform the following: ○ Place RHR A(B) in suppression pool cooling. ○ Monitor E12-R610A(B), HX A(B) OUTLET CONDUCTIVITY OR Chemistry sample from SST-PNL80, and continue flush until conductivity is less than 2 umho/cm.	MCR	3	Not required for plant shutdown or cooldown

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
o Vent RHR A(B) HX as follows: ▪ Open E12-F074A(B), RHR A(B) HX UP STREAM VENT VALVE. ▪ Open E12-F073A(B), RHR A(B) HX DN STREAM VENT VALVE. ▪ WHEN at least 1 minute has elapsed, THEN close the following: • E12-F073A(B) • E12-F074A(B) • WHEN conductivity is less than 2 umho/cm, THEN perform the following: o Secure RHR A(B) from suppression pool cooling. o Close E12-VF085A(B), LPCS FILL PUMP STOP CHECK TO RHR A DISCH (DISCH FILL PUMP STOP CHECK TO RHR B DISCH). o Continue with pump warm-up as desired.			
During cooldown, review Attachment 7, High Critical Non-Safety Related MOVs That Are Susceptible to Thermal Binding and carryout actions to prevent thermal binding or actions to unbind the valves if they are closed when the valve temperature is greater than 200°F.	MCR	3	
IF MSIVs have been closed for pressure control, THEN the following systems may be utilized as necessary to continue a cooldown at less than or equal to 100°F/hr: • RWCU system per SOP-0090, Reactor Water Cleanup System (i.e. RWCU Blowdown Operation) • Main Steam Line Drains	MCR	3	
IF/WHEN reactor pressure is less than 400 psig, THEN any running reactor feedwater pumps may be shutdown per SOP-0009, Reactor Feedwater System.	MCR / TB 67'	3	Not required for plant shutdown or cooldown
Prior to reaching 135 psig, initiate monitoring of the following parameters: • RHR Room sump levels (monitor for possible reactor vessel inventory loss from shutdown cooling leakage). (DFR-LI135 and DFR-LI138). • Suppression Pool for unexpected level rise (monitor for reactor vessel inventory loss from RHR to the Suppression Pool).	MCR	3	
WHEN RPV pressure has been lowered to below 135 psig, THEN **place one loop of RHR in Shutdown Cooling per SOP-0031, Residual Heat Removal.**	MCR	3	
RHR Pump A Warmup (SOP-0031)			
Verify closed E12-F064A, RHR PUMP A MIN FLOW TO SUP PL.	MCR	3	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Verify open E12-F047A, RHR A HX INLET VALVE.	MCR	3	
Verify closed E12-F004A, RHR PUMP A SUP PL SUCTION VALVE.	MCR	3	
Verify open E12-F006A, RHR PUMP A SDC SUCTION VALVE.	MCR	3	
On H13-P870, verify closed SPC-AOV16, SPC HX SW DISCH VLV.	MCR	3	
On H13-P870, throttle open E12-F068A, RHR HX A SVCE WTR RTN to establish less than or equal to 5800 gpm flow as indicated on E12-R602A, RHR HX A SVCE WTR FLOW.	MCR	3	
Close E12-F048A, RHR A HX BYPASS VALVE.	MCR	3	
Verify open E12-F003A, RHR A HX OUTLET VALVE.	MCR	3	
Verify open E12-F010, RHR SDC MAN ISOL VLV.	MCR	3	
In the Div 1 RSS Room at C61-PNL001, verify E12-MOVF008 ENABLE/DISABLE Switch is in ENABLE.	CB 95' Div 1 RSS Room	3	**REQUIRED**
Perform the following: • Depress B21H-S33, INBD ISOLATION SEAL-IN RESET Pushbutton. • Depress B21H-S32, OUTBD ISOLATION SEAL-IN RESET Pushbutton.	MCR	3	
At H13-P601, check RHR ISOLATION Status Lights are ON for E12-F008 and F009.	MCR	3	
Note current indicated CNS flow at LWS-PNL187 on CNS-FI116.	ACR	3	Not required for plant shutdown or cooldown
Slowly open E12-VF020, SHUTDOWN COOLING SUCTION FILL.	AB 95' RHR C Pump Room	3	Not required for plant shutdown or cooldown
WHEN CNS flow into the shutdown cooling header stops as indicated by a lack of flow noise or flow indication of approximately the same value as previously noted on CNS-FI116, THEN close E12-VF020.	AB 95' RHR C Pump Room	3	Not required for plant shutdown or cooldown
Open E12-F008, RHR SHUTDOWN COOLING OUTBD ISOL VALVE.	MCR	3	
Open E12-F009, RHR SHUTDOWN COOLING INBD ISOL VALVE.	MCR	3	
Perform STP-204-0204, RHR Shutdown Cooling Piping Fill Verification.	Steam Tunnel 114'	3	Not required for plant shutdown or cooldown
Notify Radwaste of reactor water flush to the Waste Collector Tanks.	MCR	3	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
Open E12-F049, RHR A TO RADWASTE UP STREAM ISOL VALVE.	MCR	3	
Throttle open E12-F040, RHR A TO RADWASTE DN STREAM ISOL VALVE.	MCR	3	
Monitor E12-R601, RHR TEMPERATURES, Point 1, RHR INLET TO HX1 A-1 (E12-N004A) for temperature rise and throttle E12-F040 to maintain less than or equal to 100°F/hr heatup.	MCR	3	
Continue the warm-up until E12-R601 Point 1 is within 100°F of reactor water temperature.	MCR	3	
Close E12-F049, RHR A TO RADWASTE UP STREAM ISOL VALVE.	MCR	3	
Close E12-F040, RHR A TO RADWASTE DN STREAM ISOL VALVE.	MCR	3	
Open E12-F048A, RHR HX A BYPASS VALVE.	MCR	3	
RHR Pump B Warm-up (SOP-0031)			
Verify closed E12-F064B, RHR PUMP B MIN FLOW TO SUP PL.	MCR	3	
Verify open E12-F047B, RHR B HX INLET VALVE.	MCR	3	
Verify closed E12-F004B, RHR PUMP B SUP PL SUCTION VALVE.	MCR	3	
Verify open E12-F006B, RHR PUMP B SDC SUCTION VALVE.	MCR	3	
IF Standby Service Water is supplying service water loads, THEN on H13-P870, verify closed SPC-AOV16, SPC HX SW DISCH VLV.	MCR	3	
On H13-P870, throttle open E12-F068B, RHR HX B SVCE WTR RTN to establish less than or equal to 5800 gpm flow as indicated on E12-R602B, RHR HX B SVCE WTR FLOW.	MCR	3	
Verify open E12-F010, RHR SDC MAN ISOL VLV.	MCR	3	
Verify closed E12-F049, RHR A TO RADWASTE UP STREAM ISOL VALVE.	MCR	3	
Verify closed E12-F040, RHR A TO RADWASTE DN STREAM ISOL VALVE.	MCR	3	
In the Div 1 RSS Room at C61-PNL001, verify E12-MOVF008 ENABLE/DISABLE Switch is in ENABLE.	CB 95' Div 1 RSS Room	3	**REQUIRED**
Perform the following: • Depress B21H-S32, OUTBD ISOLATION SEAL-IN RESET Pushbutton. • Depress B21H-S33, INBD ISOLATION SEAL-IN RESET Pushbutton.	MCR	3	
At H13-P601, check RHR ISOLATION Status Lights are ON for E12-F008 and F009.	MCR	3	
Note current indicated CNS flow at LWS-PNL187 on CNS-FI116.	ACR	3	Not required for plant shutdown or

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
			cooldown
Slowly open E12-VF020, SHUTDOWN COOLING SUCTION FILL.	AB 95' RHR C Pump Room	3	Not required for plant shutdown or cooldown
WHEN CNS flow into the shutdown cooling header stops as indicated by a lack of flow noise or flow indication of approximately the same value as previously noted on CNS-FI116, THEN close E12-VF020.	AB 95' RHR C Pump Room	3	Not required for plant shutdown or cooldown
Open E12-F008, RHR SHUTDOWN COOLING OUTBD ISOL VALVE.	MCR	3	
Open E12-F009, RHR SHUTDOWN COOLING INBD ISOL VALVE.	MCR	3	
Perform STP-204-0204, RHR Shutdown Cooling Piping Fill Verification.	Steam Tunnel 114'	3	Not required for plant shutdown or cooldown
Notify Radwaste of reactor water flush to the Waste Collector Tanks.	MCR	3	
Unlock and open E12-VF072B, RHR B DISCH LINE FLUSH.	AB 70' RHR B Pump Room	3	**REQUIRED**
Unlock and throttle open E12-VF070, RHR DR TO RADWASTE.	AB 80' RHR A Pump Room	3	**REQUIRED**
Monitor E12-R601, RHR TEMPERATURES, Point 11, RHR DISCH TO RADWASTE (E12-N024) and throttle E12-VF070 to maintain less than or equal to 100˚F/hr heatup.	MCR	3	
Continue the warm-up until E12-R601 Point 11 is within 100˚F of reactor water temperature.	MCR	3	
Close and lock E12-VF070, RHR DR TO RADWASTE.	AB 80' RHR A Pump Room	3	**REQUIRED**
Close and lock E12-VF072B, RHR B DISCH LINE FLUSH.	AB 70' RHR B Pump Room	3	**REQUIRED**
Startup of Shutdown Cooling (SOP-0031)			
IF any of the following RHR Shutdown Cooling interlocks are to be bypassed, THEN obtain senior plant management review and approval and verify contingency methods are in place to supply sufficient makeup water if a draining event occurs while the SDC interlocks are bypassed: • Low reactor water level isolation of E12-F008, RHR SHUTDOWN COOLING OUTBD ISOL VALVE and E12-F009, RHR SHUTDOWN COOLING INBD ISOL VALVE.	MCR	3	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
• Interlocks between E12-F004, RHR PUMP SUP PL SUCTION VALVE and E12-F006, RHR PUMP SDC SUCTION.			
On H13-P601, verify less than 135 psig Reactor Pressure as indicated on B21-R623A(B), RX LEVEL/PRESSURE RECORDER A(B).	MCR	3	
Verify closed the following: • E12-F004A(B), RHR PUMP A(B) SUP PL SUCTION VALVE • E12-F064A(B), RHR PUMP A(B) MIN FLOW TO SUP PL • E12-F024A(B), RHR PUMP A(B) TEST RTN TO SUP PL • E12-F037A(B), RHR A(B) TO UPPER POOL FPC ASSIST • E12-F048A(B), RHR A(B) HX BYPASS VALVE. • E12-F011A(B), RHR A(B) HX CNDS FLUSH TO SUP PL.	MCR	3	
Place in OFF and initiate administrative controls for EHS-MCC2E(2F) BKR 5C(7B), C002A(B) DISCH MIN FLOW VALVE.	AB 114' West	3	**REQUIRED**
IF Standby Service Water is supplying service water loads, THEN on H13-P870, verify closed SPC-AOV16, SPC HX SW DISCH VLV.	MCR	3	
On H13-P870, throttle open E12-F068A(B), RHR HX A(B) SVCE WTR RTN to establish less than or equal to 5800 gpm flow as indicated on H13-P601, E12-R602A(B), RHR HX A(B) SVCE WTR FLOW.	MCR	3	
Verify Step 4.4.2 has been performed.	MCR	3	
At H13-P601, depress B21H-S32, OUTBD ISOLATION SEAL-IN RESET Pushbutton.	MCR	3	
At H13-P601, depress B21H-S33, INBD ISOLATION SEAL-IN RESET Pushbutton.	MCR	3	
At H13-P601, check RHR ISOLATION Status Lights are ON for E12-F008 and E12-F009.	MCR	3	
In the Div 1 RSS Room at C61-PNL001, verify E12-MOVF008 ENABLE/DISABLE Switch is in ENABLE.	CB 95' Div 1 RSS Room	3	**REQUIRED**
Verify open the following: • E12-F010, RHR SDC MAN ISOL VLV • E12-F009, RHR SHUTDOWN COOLING INBD ISOL VALVE • E12-F008, RHR SHUTDOWN COOLING OUTBD ISOL VALVE • E12-F006A(B), RHR PUMP A(B) SDC SUCTION VALVE • E12-F047A(B), RHR A(B) HX INLET VALVE	MCR	3	
Verify open one of the following: • E12-F053A(B), RHR PUMP A(B) SDC INJECTION VALVE • E12-F037A(B), RHR A(B) TO UPPER POOL FPC ASSIST	MCR	3	
Close E12-F003A(B), RHR A(B) HX OUTLET VALVE.	MCR	3	
Start E12-C002A(B), RHR PUMP A(B) and IMMEDIATELY throttle open E12-F048A(B), RHR A(B) HX BYPASS VALVE to obtain greater	MCR	3	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
than or equal to 2000 gpm and less than or equal to 3000 gpm.			
Establish a stable flow of greater than or equal to 4000 gpm and less than or equal to 5000 gpm by throttling E12-F048A(B), RHR A(B) HX BYPASS VALVE.	MCR	3	
Throttle open E12-F003A(B), RHR A(B) HX OUTLET VALVE to approximately 10 PERCENT as indicated on E12-R611A(B), HX A(B) OUTLET VLV POS.	MCR	3	
Establish a cooldown rate of less than 100°F/hr as follows: • Slowly throttle open E12-F003A(B) RHR A(B) HX OUTLET VALVE and monitor the cooldown rate. • Throttle E12-F003A(B), RHR A(B) HX OUTLET VALVE and E12-F048A(B), RHR A(B) HX BYPASS VALVE to obtain the desired cooldown rate or maintain the desired coolant temperature while maintaining a constant RHR loop flow. • IF shifting divisions of Shutdown Cooling per Section 5.6, THEN in the other RHR loop, throttle E12-F003B(A), RHR B(A) HX OUTLET VALVE and E12-F048B(A), RHR B(A) HX BYPASS VALVE to maintain the desired cooldown rate or coolant temperature while maintaining a constant RHR loop flow. • Close FWS-MOV7A(B), A(B) FW OUTBD ISOL.	MCR	3	
IF it is desired to establish RHR Shutdown Cooling Protection, THEN Go To Section 4.5.	MCR / AB 95' & 115'		Not required for plant shutdown or cooldown
WHEN RHR Shutdown Cooling is established and adequate RPV makeup is assured via CRD or Feedwater, THEN close FWS-MOV7A(B), A(B) FW OUTBD ISOL valve on the Feedwater Header supporting RHR Shutdown Cooling.	MCR	3	
WHEN RPV cooldown is being conducted using RHR Shutdown Cooling, THEN stop discharging steam to the main condenser, break condenser vacuum and continue to shutdown the turbine plant as follows: • Place CONDENSER LOW VACUUM BYPASS Switches to BYPASS. • Close all turbine bypass valves and steam drain valves. • Open CNM-AOVVB, CNDS VAC BRKR. • WHEN condenser vacuum reaches approximately 0" Hg, THEN shutdown steam seals per SOP-0015, Gland Seal System and Exhaust System.	MCR	3	
WHEN condenser vacuum reaches 0" Hg AND mechanical vacuum pump operation is no longer required, THEN align Alternate Hotwell Level tygon tubing per SOP-0008, Condensate Storage, Makeup and Transfer.	TB 67'	3	Not required for plant shutdown or cooldown

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
At less than 190°F, perform the following: • Open B21-MOVF001, RX DN STREAM HEAD VENT TO DW EQPT DR SUMP. • Open B21-MOVF002, RX UP STREAM HEAD VENT TO DW EQPT DR SUMP. • Close B21-MOVF005, RX HEAD VENT TO MSL A. • Establish administrative controls to maintain the established vessel vent path until vessel head piping is disassembled.	MCR	3	
IF required/desired, THEN close the MSIVs by placing the following in CLOSE: • B21-F028B, MSL B OUTBD MSIV • B21-F028D, MSL D OUTBD MSIV • B21-F028A, MSL A OUTBD MSIV • B21-F028C, MSL C OUTBD MSIV • B21-F022B, MSL B INBD MSIV • B21-F022D, MSL D INBD MSIV • B21-F022A, MSL A INBD MSIV • B21-F022C, MSL C INBD MSIV	MCR	3	
At less than 200°F, Mode 4, perform the following: At H13-P632, place the following switches to BYPASS: • E31A-S1A, RWCU ISOLATION BYPASS DIV 1 • E31A-S2A, RCIC ISOLATION BYPASS DIV 1 • E31A-S4A, RHR ISOLATION BYPASS DIV 1	MCR	3	
At H13-P642, place the following switches to BYPASS: • E31A-S1B, RWCU ISOLATION BYPASS DIV 2 • E31A-S2B, RCIC ISOLATION BYPASS DIV 2 • E31A-S4B, RHR ISOLATION BYPASS DIV 2	MCR	3	
Implement Shutdown Cooling Protection per SOP-0031, Residual Heat Removal.	MCR / AB 95' & 115'	3	Not required for plant shutdown or cooldown
Bypass RPS trip logic using EOP-0005 Enclosure 12 Bypass Switches per SOP-0079, Reactor Protective System.	MCR	3	
Bypass ARI logic trips per SOP-0079.	MCR	3	
Bypass Backup Scram Valve trips per SOP-0079.	MCR	3	
At less than 200°F, Mode 4, perform the following to prevent isolating Breathing Air: • Verify open SAS-MOV102, SVCE AIR OUTBD ISOL	MCR	3	Not required for plant shutdown or

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
			cooldown
• Open EHS-MCC2J Bkr 3C, SAS-MOV102 CONTAINMENT SERVICE AIR ISOLATION.	AB 141' West	3	Not required for plant shutdown or cooldown
• Hang the following SAS-MOV102 Caution Tags stating that before closing the breaker or valve, verify that Breathing Air is not in use.: o EHS-MCC2J Bkr 3C o SAS-MOV102 local handwheel o SAS-MOV102 MCR control switch	MCR	3	Not required for plant shutdown or cooldown
• Hang "Breathing Air in Use" sign in the MCR.	MCR	3	Not required for plant shutdown or cooldown
• Make an "Open Item" Narrative Log entry by checking the "Open Item" box stating that "Breathing Air Is in Use" to carry over until Breathing Air is no longer in use.	MCR	3	Not required for plant shutdown or cooldown
At less than 200°F, notify Chemistry to consider securing the Durability Monitor.	AB 114' Crescent Area	3	Not required for plant shutdown or cooldown
IF required/desired, THEN shutdown Reactor Recirculation System per SOP-0003, Reactor Recirculation System and raise reactor water level to at least 75 inches on shutdown range level instrumentation.	MCR	3	Not required for plant shutdown or cooldown
As necessary, reduce the number of operating Turbine Building Chillers per SOP-0064 to prevent the chillers from tripping on low load.	TB 67'	3	Not required for plant shutdown or cooldown
GOP-0003 Scram Recovery			
Verify/establish on-scale neutron monitoring on the SRMs and IRMs • VERIFY the SRM Channel Functional Tests are current. If Channel Functional Tests are not current, refer to Tech Spec 3.3.1.2.	MCR	3	
Maintain RPV pressure to prevent excessive cooldown rates or RPV overpressurization by: • Use of Main Turbine Bypass System. • Use of Normal Plant Steam Loads/Steam Line Drains. • RCIC in CST to CST Mode per SOP-0035 Reactor Core Isolation Cooling System.	MCR	3	

Attachment 2 – Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
• Alternate opening of SRVs as needed (only on RPV isolation).			
Maintain RPV Level using: • Condensate/Feedwater • RCIC • HPCS • CRD/RWCU	MCR	3	
If tripped, restart Reactor Recirculation Pumps on LFMG per SOP-0003, Reactor Recirculation: • Open flow control valves to the full open position.	MCR / LFMG Room	3	Not required for plant shutdown or cooldown
Verify Main Turbine steam seals are being maintained at approximately 4 psig.	MCR	3	
Start air removal pumps per SOP-0025 Condenser Air Removal System. • Maintain condenser vacuum between 23" Hg and 28" Hg.	MCR / TB 67'	3	Not required for plant shutdown or cooldown
IF the Steam Jet Air Ejectors have been lost, THEN Secure Offgas System per SOP-0092, to establish purge air flow in order to prevent system reverse flow.	MCR / TB 95' & 123'	3	Not required for plant shutdown or cooldown
Notify Chemistry Department to operate the Offgas Hydrogen Analyzers per COP-0227, Operation of the Offgas Hydrogen Analyzers.	MCR / TB 123'	3	Not required for plant shutdown or cooldown
Record the highest vessel pressure indicated by tracking pointer on B21-PIR004A and B21-PIR004B (114' Containment). Reset the tracking pointer.	RB 114'	3	Not required for plant shutdown or cooldown
Inspect all CRD HCUs for leakage due to piping cracks. • Notify Engineering NDE that visual inspections of HCU charging water piping are required.	RB 114'	3	Not required for plant shutdown or cooldown
WHEN FWREG Valves are removed from service, THEN perform SOP-0009 Attachment for Calibration Check of FWREG Valve.	MCR / TB 67'	3	Not required for plant shutdown or cooldown
Go To GOP-0002 - Start at the beginning of GOP-0002 and complete	MCR	3	

GOP / SOP ACTIONS	LOCATION	MODE	NOTES
all steps required to place the plant in the desired mode after scram.			
IF the plant tripped while connected to the grid, THEN notify Site Design Engineering to notify the TOP personnel of the event per ENS-DC-201, ENS Transmission Grid Monitoring Attachment 9.3 Step 3.0[1].	MCR	3	
Engineering perform a review of post scram cooldown data and compare to PT Curves provided in STP-050-0700 Attachment 3. Also verify the cooldown rate is bounded by analyzed thermal cycles.	MCR	3	
Shift Manager to perform a Post SCRAM crew critique identifying all Human Performance issues and Equipment Malfunctions. Document each item on separate CRs. Attach Crew Critique to this procedure.	MCR	3	
AOM/Shift Manager review equipment malfunctions and recommend to OSRC required repairs prior to restart. This should include an evaluation of risk mitigating Non TRM structures, systems, and components.	MCR	3	

Control Room ventilation systems have adequate engineered safety/design features in place to preclude a Control Room evacuation due to the release of a hazardous gas. Therefore, the Control Room is not included in this assessment or in Table H-2.

Table A-3 & H-2 Results

Table A-3 & H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode**
Auxiliary Building 70' RHR B Pump Room	3
Auxiliary Building 80' RHR A Pump Room	3
Auxiliary Building 114' West	3
Control Building 95' Div 1 RSS Room	3

Mode 3 is included above for SDC-related activities because the procedures begin alignment in Mode 3; however, these actions could be delayed until Mode 4, if necessary. In order to ensure adequate guidance to emergency response personnel, the above areas are added to the EAL in order to provide prompt operator guidance for EAL declaration.

Control Room ventilation systems have adequate engineered safety/design features in place to preclude a Control Room evacuation due to the release of a hazardous gas. Therefore, the Control Room is not included in this assessment or in Table H-2.

ENCLOSURE 4
RBG-47847

NEI 99-01, REV. 6, DEVIATIONS AND DIFFERENCES, RBS

RBS

NEI 99-01 Revision 6

EAL Comparison Matrix

Table of Contents

Introduction

This document provides a line-by-line comparison of the Initiating Conditions (ICs), Mode Applicability and Emergency Action Levels (EALs) in NEI 99-01 Rev. 6 Final, Development of Emergency Action Levels for Non-Passive Reactors, ADAMS Accession Number ML12326A805, and River Bend Station (RBS) ICs, Mode Applicability and EALs. This document provides a means of assessing RBS differences and deviations from the NRC endorsed guidance given in NEI 99-01. Discussion of RBS EAL bases and lists of source document references are given in the EAL Technical Bases Document. It is, therefore, advisable to reference the EAL Technical Bases Document for background information while using this document.

Comparison Matrix Format

The ICs and EALs discussed in this document are grouped according to NEI 99-01 Recognition Categories. Within each Recognition Category, the ICs and EALs are listed in tabular format according to the order in which they are given in NEI 99-01. Generally, each row of the comparison matrix provides the following information:

- NEI EAL/IC identifier
- NEI EAL/IC wording
- RBS EAL/IC identifier
- RBS EAL/IC wording
- Description of any differences or deviations

EAL Wording

In Section 4.1, NEI recommends the following: "The guidance in NEI 99-01 is not intended to be applied to plants "as-is"; however, developers should attempt to keep their site-specific schemes as close to the generic guidance as possible. The goal is to meet the intent of the generic Initiating Conditions (ICs) and Emergency Action Levels (EALs) within the context of site-specific characteristics – locale, plant design, operating features, terminology, etc. Meeting this goal will result in a shorter and less cumbersome NRC review and approval process, closer alignment with the schemes of other nuclear power plant sites and better positioning to adopt future industry-wide scheme enhancements"

To assist the Emergency Director (ED), the RBS EALs have been written in a clear and concise style (to the extent that the differences from the NEI EAL wording could be reasonably documented and justified). This supports timely and accurate classification in the tense atmosphere of an emergency event. The EAL differences introduced to reduce reading burden comprise almost all of the differences justified in this document.

EAL Emphasis Techniques

Due to the width of the table columns and table formatting constraints in this document, line breaks and indentation may differ slightly from the appearance of comparable wording in the source documents. NEI 99-01 is the source document for the NEI EALs; the RBS EAL Technical Bases Document for the RBS EALs.

Development of the RBS IC/EAL wording has attempted to minimize inconsistencies and apply sound human factors principles. As a result, differences occur between NEI and RBS ICs/EALs for these reasons alone. When such difference may infer a technical difference in the associated NEI IC/EAL, the difference is identified and a justification provided.

The print and paragraph formatting conventions summarized below guide presentation of the RBS EALs in accordance with the EAL writing criteria. Space restrictions in the EAL table of this document sometimes override this criteria in cases when following the criteria would introduce undesirable complications in the EAL layout.

- Upper case-bold print is used for the logic terms **AND**, **OR** and **EITHER**.
- Bold font is used for certain logic terms, negative terms (**not**, **cannot**, etc.), **any**, **all**.
- Upper case print is reserved for defined terms, acronyms, system abbreviations, logic terms (and, or, etc. when not used as a conjunction), annunciator window engravings.
- Three or more items in a list are normally introduced with "**Any** of the following..." or "**All** of the following..." Items of the list begin with bullets when a priority or sequence is not inferred.
- The use of **AND/OR** logic within the same EAL has been avoided when possible. When such logic cannot be avoided, indentation and separation of subordinate contingent phrases is employed.

Global Differences

The differences listed below generally apply throughout the set of EALs and are not repeated in the Justification sections of this document. The global differences do not decrease the effectiveness of the intent of NEI 99-01.

1. The NEI phrase "Notification of Unusual Event" has been changed to "Unusual Event" or abbreviated "UE" to reduce EAL-user reading burden.
2. NEI 99-01 IC Example EALs are implemented in separate plant EALs to improve clarity and readability. For example, NEI lists all IC HU3 Example EALs under one IC. The corresponding RBS 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 Shutdown, 4 - Cold Shutdown, 5 - Refueling, DEF – Defueled. NEI 99-01defines Defueled as follows: "Reactor Vessel contains no irradiated fuel (full core off-load during refueling or extended outage)."
4. The Hot Standby mode is applicable only to PWRs and therefore is not used for RBS.
5. NEI 99-01 uses the terms greater than, less than, greater than or equal to, etc. in the wording of some example EALs. For consistency and reduce EAL-user reading burden, RBS has adopted use of Boolean symbols in place of the NEI 99-01 text modifiers within the EAL wording.
6. "min." is the standard abbreviation for "minutes" and is used to reduce EAL user reading burden.
7. The terms "increase" and "decrease" have been replaced with the terms "rise" and "lower".
8. IC/EAL identification:
   - NEI 99-01 defines the thresholds requiring emergency classification (example EALs) and assigns them to ICs which, in turn, are grouped in "Recognition Categories." RBS endeavors to optimize the NEI EAL organization and identification scheme to enhance usability of the plant-specific EAL set. To this end, the RBS 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, 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 RBS EAL categories/subcategories and their relationship to NEI Recognition Categories are listed in Table 1.

c. Unique identification of each EAL – Four characters comprise the EAL identifier as illustrated in Figure 1.

Figure 1 – EAL Identifier

EAL Identifier

XXX.X

Category (A, H, E, S, F, C)

Emergency classification (G, S, A, U)

Sequential number within subcategory/classification

Subcategory number (1 if no subcategory)

The first character is a letter associated with the category in which the EAL is located. The second character is a letter associated with the emergency classification level (G for General Emergency, S for Site Area Emergency, A for Alert, and U for Notification of Unusual Event). The third character is a number associated with one or more subcategories within a given category. Subcategories are sequentially numbered beginning with the number "1". If a category does not have a subcategory, this character is assigned the number "1". The fourth character is a number preceded by a period for each EAL within a subcategory. EALs are sequentially numbered within the emergency classification level of a subcategory beginning with the number "1".

The EAL identifier is designed to fulfill the following objectives:

- Uniqueness – The EAL identifier ensures that there can be no confusion over which EAL is driving the need for emergency classification.
- Speed in locating the EAL of concern – When the EALs are displayed in a matrix format, knowledge of the EAL identifier alone can lead the EAL-user to the location of the EAL within the classification matrix. The identifier conveys the category, subcategory and classification level. This assists ERO responders (who may not be in the same facility as the ED) to find the EAL of concern in a timely manner without the need for a word description of the classification threshold.
- Possible classification upgrade – The category/subcategory/identifier scheme helps the EAL-user find higher emergency classification EALs that may become active if plant conditions worsen.

Table 2 lists the RBS 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 RBS 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 RBS 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 RBS (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 RBS equipment/instrument identification and/or noun names to EALs.
- Combining like ICs that are exactly the same but have different operating modes as long as the intent of each IC is maintained and the overall progression of the EAL scheme is not affected.
- Any change to the IC and/or EAL, and/or basis wording, as stated in NEI 99-01, that does not alter the intent of the IC and/or EAL, i.e., the IC and/or EAL continues to:
    - Classify at the correct classification level.
    - Logically integrate with other EALs in the EAL scheme.
    - Ensure that the resulting EAL scheme is complete (i.e., classifies all potential emergency conditions).

The following are examples of deviations:

- Use of altered mode applicability.
- Altering key words or time limits.
- Changing words of physical reference (protected area, safety-related equipment, etc.).
- Eliminating an IC. This includes the removal of an IC from the Fission Product Barrier Degradation category as this impacts the logic of Fission Product Barrier ICs.
- Changing a Fission Product Barrier from a Loss to a Potential Loss or vice-versa.
- Not using NEI 99-01definitions as the intent is for all NEI 99-01 users to have a standard set of defined terms as defined in NEI 99-01. Differences due to plant types are permissible (BWR or PWR). Verbatim compliance to the wording in NEI 99-01 is not necessary as long as the intent of the defined word is maintained. Use of the wording provided in NEI 99-01 is encouraged since the intent is for all users to have a standard set of defined terms as defined in NEI 99-01.
- Any change to the IC and/or EAL, and/or basis wording as stated in NEI 99-01 that does alter the intent of the IC and/or EAL, i.e., the IC and/or EAL:
    - Does not classify at the classification level consistent with NEI 99-01.
    - Is not logically integrated with other EALs in the EAL scheme.
    - Results in an incomplete EAL scheme (i.e., does not classify all potential emergency conditions).

The "Difference/Deviation Justification" columns in the remaining sections of this document identify each difference between the NEI 99-01 IC/EAL wording and the RBS IC/EAL wording. An explanation that justifies the reason for each difference is then provided. If the difference is determined to be a deviation, a statement is made to that affect and explanation is given that states why classification may be different from the NEI 99-01 IC/EAL and the reason for its acceptability. In all cases, however, the differences and deviations do not decrease the effectiveness of the intent of NEI 99-01. A summary list of RBS EAL deviations from NEI 99-01 is given in Table 3.

Table 1 – RBS EAL Categories/Subcategories

RBS EALs		NEI Recognition Category
Category	**Subcategory**	
Group: Any Operating Mode:		
A – 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 – Emergency Director Judgment	**H**azards and Other Conditions Affecting Plant Safety ICs/EALs
E - ISFSI	1 – Confinement Boundary	ISFSI ICs/EALs
Group: Hot Conditions:		
S – System Malfunction	1 – Loss of Emergency AC Power 2 – Loss of Vital DC Power 3 – Loss of Control Room Indications 4 – RCS Activity 5 – RCS Leakage 6 – RPS Failure 7 – Loss of Communications 8 – Hazardous Event Affecting Safety Systems	**S**ystem Malfunction ICs/EALs
F – Fission Product Barrier	None	**F**ission Product Barrier ICs/EALs
Group: Cold Conditions:		
C – Cold Shutdown/Refueling System Malfunction	1 – RPV Level 2 – Loss of Emergency AC Power 3 – RCS Temperature 4 – Loss of Vital DC Power 5 – Loss of Communications 6 - Hazardous Event Affecting Safety Systems	**C**old Shutdown./ Refueling System Malfunction ICs/EALs

Table 2 – NEI / RBS EAL Identification Cross-Reference

NEI		RBS	
IC	Example EAL	Category and Subcategory	EAL
AU1	1	A – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	AU1.1
AU1	2	A – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	AU1.1
AU1	3	A – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	AU1.2
AU2	1	A – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	AU2.1
AA1	1	A – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	AA1.1
AA1	2	A – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	AA1.2
AA1	3	A – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	AA1.3
AA1	4	A – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	AA1.4
AA2	1	A – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	AA2.1
AA2	2	A – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	AA2.2
AA2	3	A – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	AA2.3
AA3	1	A – Abnormal Rad Levels / Rad Effluent, 3 – Area Radiation Levels	AA3.1
AA3	2	A – Abnormal Rad Levels / Rad Effluent, 3 – Area Radiation Levels	AA3.2
AS1	1	A – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	AS1.1
AS1	2	A – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	AS1.2
AS1	3	A – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	AS1.3

NEI		RBS	
IC	Example EAL	Category and Subcategory	EAL
AS2	1	A – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	AS2.1
AG1	1	A – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	AG1.1
AG1	2	A – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	AG1.2
AG1	3	A – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent	AG1.3
AG2	1	A – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event	AG2.1
CU1	1	C – Cold SD/ Refueling System Malfunction, 1 – RPV Level	CU1.1
CU1	2	C – Cold SD/ Refueling System Malfunction, 1 – RPV Level	CU1.2
CU2	1	C – Cold SD/ Refueling System Malfunction, 2 – Loss of Emergency AC Power	CU2.1
CU3	1	C – Cold SD/ Refueling System Malfunction, 3 – RCS Temperature	CU3.1
CU3	2	C – Cold SD/ Refueling System Malfunction, 3 – RCS Temperature	CU3.2
CU4	1	C – Cold SD/ Refueling System Malfunction, 4 – Loss of Vital DC Power	CU4.1
CU5	1, 2, 3	C – Cold SD/ Refueling System Malfunction, 5 – Loss of Communications	CU5.1
CA1	1	C – Cold SD/ Refueling System Malfunction, 1 – RPV Level	CA1.1
CA1	2	C – Cold SD/ Refueling System Malfunction, 1 – RPV Level	CA1.2
CA2	1	C – Cold SD/ Refueling System Malfunction, 1 – Loss of Emergency AC Power	CA2.1
CA3	1, 2	C – Cold SD/ Refueling System Malfunction, 3 – RCS Temperature	CA3.1
CA6	1	C – Cold SD/ Refueling System Malfunction, 6 – Hazardous Event Affecting Safety Systems	CA6.1
CS1	1	C – Cold SD/ Refueling System Malfunction, 1 – RPV Level	CS1.1

NEI		RBS	
IC	Example EAL	Category and Subcategory	EAL
CS1	2	C – Cold SD/ Refueling System Malfunction, 1 – RPV Level	CS1.2
CS1	3	C – Cold SD/ Refueling System Malfunction, 1 – RPV Level	CS1.3
CG1	1	C – Cold SD/ Refueling System Malfunction, 1 – RPV Level	CG1.1
CG1	2	C – Cold SD/ Refueling System Malfunction, 1 – RPV Level	CG1.2
E-HU1	1	E - ISFSI	EU1.1
FA1	1	F – Fission Product Barrier Degradation	FA1.1
FS1	1	F – Fission Product Barrier Degradation	FS1.1
FG1	1	F – Fission Product Barrier Degradation	FG1.1
HU1	1, 2, 3	H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security	HU1.1
HU2	1	H – Hazards and Other Conditions Affecting Plant Safety, 2 – Seismic Event	HU2.1
HU3	1	H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technology Hazard	HU3.1
HU3	2	H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technology Hazard	HU3.2
HU3	3	H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technology Hazard	HU3.3
HU3	4	H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technology Hazard	HU3.4
HU3	5	H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technology Hazard	N/A
HU4	1	H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire or Explosion	HU4.1
HU4	2	H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire or Explosion	HU4.2
HU4	3	H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire or Explosion	HU4.3

NEI		RBS	
IC	Example EAL	Category and Subcategory	EAL
HU4	4	H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire or Explosion	HU4.4
HU7	1	H – Hazards and Other Conditions Affecting Plant Safety, 7 – Judgment	HU7.1
HA1	1, 2	H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security	HA1.1
HA5	1	H – Hazards and Other Conditions Affecting Plant Safety, 5 – Hazardous Gases	HA5.1
HA6	1	H – Hazards and Other Conditions Affecting Plant Safety, 6 – Control Room Evacuation	HA6.1
HA7	1	H – Hazards and Other Conditions Affecting Plant Safety, 7 – Judgment	HA7.1
HS1	1	H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security	HS1.1
HS6	1	H – Hazards and Other Conditions Affecting Plant Safety, 6 – Control Room Evacuation	HS6.1
HS7	1	H – Hazards and Other Conditions Affecting Plant Safety, 7 – Judgment	HS7.1
HG1	1	N/A	N/A
HG7	1	H – Hazards and Other Conditions Affecting Plant Safety, 7 – Judgment	HG7.1
SU1	1	S – System Malfunction, 1 – Loss of Emergency AC Power	SU1.1
SU2	1	S – System Malfunction, 3 – Loss of Control Room Indications	SU3.1
SU3	1	S – System Malfunction, 4 – RCS Activity	SU4.1
SU3	2	S – System Malfunction, 4 – RCS Activity	SU4.2
SU4	1, 2, 3	S – System Malfunction, 5 – RCS Leakage	SU5.1
SU5	1	S – System Malfunction, 6 – RPS Failure	SU6.1
SU5	2	S – System Malfunction, 6 – RPS Failure	SU6.2

NEI		RBS	
IC	Example EAL	Category and Subcategory	EAL
SU6	1, 2, 3	S – System Malfunction, 7 –Loss of Communications	SU7.1
SU7	1, 2	S – System Malfunction, 8 –Containment Failure	N/A
SA1	1	S – System Malfunction, 1 – Loss of Emergency AC Power	SA1.1
SA2	1	S – System Malfunction, 3 – Loss of Control Room Indications	SA3.1
SA5	1	S – System Malfunction, 6 – RPS Failure	SA6.1
SA9	1	S – Hazardous Event Affecting Safety Systems	SA8.1
SS1	1	S – System Malfunction, 1 – Loss of Emergency AC Power	SS1.1
SS5	1	S – System Malfunction, 6 – RPS Failure	SS6.1
SS8	1	S – System Malfunction, 2 – Loss of Vital DC Power	SS2.1
SG1	1	S – System Malfunction, 1 – Loss of Emergency AC Power	SG1.1
SG8	1	S – System Malfunction, 1 – Loss of Emergency AC Power	SG1.2

Table 3 – Summary of Deviations

NEI		RBS EAL	Description
IC	Example EAL		
FA1, FS1	RCS 3.A Potential Loss	RCB4	River Bend uses Technical Specification based isolation temperatures vice Max Normal Operating temperatures for NEI RCS barrier potential loss 3.A.1. A valid indication of area temperature(s) greater than or equal to the system MOV Technical Requirements Manual (TRM) isolation value resulting from a primary system discharging into the Auxiliary Building is indicative of conditions in which significant RCS inventory is being lost. This is therefore considered to be a potential loss of the RCS barrier. The Maximum Normal Operating area radiation values are still being used by River Bend and are consistent with the NEI guidance. The alarms for the high area ambient temperature are associated with the TRM 3.3.6.1 values for primary containment isolation. The use of the TRM value provides a readily identifiable condition of the RCS barrier status with an alarm in the Control Room and an isolation signal that (if the condition persists) indicates unisolable leakage as defined by the EAL threshold. The Technical Specification 3.3.6.1 ambient temperature monitoring Allowable Value setpoints are set low enough to detect a leak equivalent to 25 gpm.

Primary Containment Operating Values - Temperatures				
Parameter	Max Normal	TRM Isolation Value	TS Allowable Value	Max Safe
Main Steam Line Tunnel	144°F	173°F	183°F	200°F
RHR Equip Area A	110°F	117°F	121.1°F	200°F
RHR Equip Area B	110°F	117°F	186.4°F	200°F
RCIC Equip Area	144°F	182°F	186.4°F	200°F
RWCU Pump Room	145°F	165°F	169.5°F	200°F

<table>
<tr><th colspan="2">NEI</th><th rowspan="2">RBS EAL</th><th rowspan="2">Description</th></tr>
<tr><th>IC</th><th>Example EAL</th></tr>
<tr><td></td><td></td><td></td><td>
<table>
<tr><th colspan="3">Technical Specification 3.3.6.1 Bases – Isolation References</th></tr>
<tr><th>Area</th><th>Allowable Value</th><th>Section</th></tr>
<tr><td>Main Steam Line Isolation</td><td>183°F</td><td>1.e</td></tr>
<tr><td>RHR System Isolation</td><td>121.1°F</td><td>5.a.</td></tr>
<tr><td>RCIC Isolation</td><td>186.4°F</td><td>3.e.</td></tr>
<tr><td>RWCU Isolation</td><td>169.5°F</td><td>4.d.</td></tr>
</table>
The use of the isolation alarm and EOP condition provides a method of rapid identification of the EAL threshold without the task of performing additional confirmatory actions. The use of isolation temperatures still allows for a discernable margin prior to reaching the Max Safe Operating temperature described in NEI Containment barrier loss threshold 3.C.
Therefore this is an acceptable deviation from the generic NEI 99-01 Revision 6 guidance.
</td></tr>
<tr><td>HG1</td><td>1</td><td>N/A</td><td>
IC HG1 and associated example EAL is not implemented in the RBS scheme.
There are several other ICs that are redundant with this IC, and are better suited to ensure timely and effective emergency declarations. In addition, the development of new spent fuel pool level EALs, as a result of NRC Order EA-12-051, clarified the intended emergency classification level for spent fuel pool level events. This deviation is justified because:
1. Hostile Action in the Protected Area is bounded by ICs HS1 and HS7. Hostile Action resulting in a loss of physical control is bound by EAL HG7, as well as any event that may lead to radiological releases to the public in excess of Environmental Protection Agency (EPA) Protective Action Guides (PAGs).
a. If, for whatever reason, the Control Room must be evacuated, and control of safety functions (e.g., reactivity control, core cooling, and RCS heat removal) cannot be reestablished, then IC HS6 would apply, as well as IC HS7 if desired by the EAL decision-maker.
b. Also, as stated above, any event (including Hostile Action) that could reasonably be expected to have a release exceeding EPA PAGs would be
</td></tr>
</table>

NEI		RBS EAL	Description
IC	Example EAL		
			bound by IC HG7. c. From a Hostile Action perspective, ICs HS1, HS7 and HG7 are appropriate, and therefore, make this part of HG1 redundant and unnecessary. d. From a loss of physical control perspective, ICs HS6, HS7 and HG7 are appropriate, and therefore, make this part of HG1 redundant and unnecessary. 2. Any event which causes a loss of spent fuel pool level will be bounded by ICs AA2, AS2 and AG2, regardless of whether it was based upon a Hostile Action or not, thus making this part of HG1 redundant and unnecessary. a. An event that leads to a radiological release will be bounded by ICs AU1, AA1, AS1 and AG1. Events that lead to radiological releases in excess of EPA PAGs will be bounded by EALs AG1 and HG7, thus making this part of HG1 redundant and unnecessary. ICs AA2, AS2, AG2, AS1, AG1, HS1, HS6, HS7 and HG7 have been implemented consistent with NEI 99-01 Revision 6 and thus HG1 is adequately bounded as described above. **Therefore this is an acceptable deviation from the generic NEI 99-01 Revision 6 guidance and is consistent with NRC-approved EP FAQ 2015-013.**
HS6	1	HS6.1	Deleted defueled mode applicability. Control of the cited safety functions are not critical for a defueled reactor as there is no energy source in the reactor vessel or RCS. The Mode applicability for the reactivity control safety function has been limited to Modes 1, 2, and 3 (hot operating conditions). In the cold operating modes adequate shutdown margin exists under all conditions. **Therefore this is an acceptable deviation from the generic NEI 99-01 Revision 6 guidance and is consistent with NRC-approved EP FAQ 2015-014.**

NEI		RBS EAL	Description
IC	Example EAL		
CA6 SA9	1 1	CA6.1 SA8.1	The proposed RBS CA6.1 and SA8.1 wording is intended to ensure that an Alert should be declared only when actual or potential performance issues with SAFETY SYSTEMS have occurred as a result of a hazardous event. The occurrence of a hazardous event will result in an Unusual Event classification at a minimum. In order to warrant escalation to the Alert classification, the hazardous event should cause indications of degraded performance to one train of a SAFETY SYSTEM with either indications of degraded performance on the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second SAFETY SYSTEM train, such that the operability or reliability of the second train is a concern. In addition, escalation to the Alert classification should not occur if the damage from the hazardous event is limited to a SAFETY SYSTEM that was inoperable, or out of service, prior to the event occurring. As such, the proposed EALs will reduce the potential of declaring an Alert when events are in progress that do not involve an actual or potential substantial degradation of the level of safety of the plant, i.e., does not cause significant concern with shutting down or cooling down the plant. EALs CA6.1 and SA8.1 do not directly escalate to a Site Area Emergency or a General Emergency due to a hazardous event. The Fission Product Barrier and/or Abnormal Radiation Levels/Radiological Effluent recognition categories would provide an escalation path to a Site Area Emergency or a General Emergency. The EALs and the Basis sections have been revised to ensure potential escalations from an Unusual Event to an Alert, due to a hazardous event, is appropriate as the concern with these EALs is: (1) a hazardous event has occurred, (2) one SAFETY SYSTEM train is having performance issues as a result of the hazardous event, and (3) either the second SAFETY SYSTEM train is having performance issues or the VISIBLE DAMAGE is enough to be concerned that the second SAFETY SYSTEM train may have operability or reliability issues. The definition for VISIBLE DAMAGE has been revised to reflect the fact that the EALs are based upon SAFETY SYSTEM trains rather than individual components or structures. Note 9 has been added to CA6.1 and SA8.1 as it meets the intent of the EALs, is consistent with other EALs (e.g., EAL HA5.1 which was previously endorsed by the NRC), and ensures that declared emergencies are based upon unplanned

NEI		RBS EAL	Description
IC	Example EAL		
			events with the potential to pose a radiological risk to the public. Note 10 has been added to CA6.1 and SA8.1 to help reinforce and succinctly capture the more detailed information from the revised basis section related to when conditions would require the declaration of an Alert. CA6.1 and SA8.1 are consistent with approved NRC FAQ 2016-002 addressing degraded performance or visible damage to more than one safety system train caused by the specified events. **Based on the above information, this revised wording is an acceptable deviation from the generic NEI 99-01 Revision 6 guidance and is consistent with NRC approved EP FAQ 2016-002.**

Category A

Abnormal Rad Levels / Radiological Effluent

NEI IC#	NEI IC Wording and Mode Applicability	RBS IC#(s)	RBS IC Wording and Mode Applicability	Difference/Deviation Justification
AU1	Release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer. MODE: All	AU1	Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer MODE: All	The RBS ODCM is the site-specific effluent release controlling document.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Reading on **ANY** effluent radiation monitor greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer: (site-specific monitor list and threshold values corresponding to 2 times the controlling document limits)	AU1.1	Reading on **any** Table A-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3)	Example EALs #1 and #2 have been combined into a single EAL to simplify presentation. The NEI phrase "...effluent radiation monitor greater than 2 times the (site-specific effluent release controlling document)" and "effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit " have been replaced with "...**any** Table A-1 effluent radiation monitor > column "UE". UE thresholds for all RBS continuously monitored gaseous and liquid release pathways are listed in Table A-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL user. The values shown in Table A-1 column "UE", consistent with the NEI bases, represent two times the ODCM release limits for gaseous and liquid releases.
2	Reading on **ANY** effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.			
3	Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the (site-specific effluent release controlling document) limits for	AU1.2	Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2)	The RBS ODCM is the site-specific effluent release controlling document.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
	60 minutes or longer.			
Notes	• The Emergency Director should declare the Unusual Event promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded. • If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 60 minutes. • If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded. Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit. Note 3: If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer VALID for classification purposes.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock. The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. None

Table A-1 Effluent Monitor Classification Thresholds						
Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	Main Plant Vent - Primary	RE125	9.56E+08 µCi/sec	9.56E+07 µCi/sec	9.63E+06 µCi/sec	1.01E+05 µCi/sec
	Main Plant Vent - Secondary	RE126	----	----	1.66E-01 µCi/ml	1.74E-03 µCi/ml
	Fuel Bldg Vent - Primary	RE5A	7.75E+08 µCi/sec	7.75E+07 µCi/sec	7.75E+06 µCi/sec	6.50E+03 µCi/sec
	Fuel Bldg Vent - Secondary	RE5B	----	----	1.72E-01µCi/ml	1.38E-03 µCi/ml
	Radwaste Bldg Vent - Primary	RE6A	8.03E+08 µCi/sec	8.03E+07 µCi/sec	8.03E+06 µCi/sec	6.96E+04 µCi/sec
	Radwaste Bldg Vent - Secondary	RE6B	----	----	2.12E-01 µCi/ml	1.71E-04 µCi/ml
Liquid	Liquid Radwaste	RE107	----	----	----	2 x Alarm Setpoint

NEI IC#	NEI IC Wording and Mode Applicability	RBS IC#(s)	RBS IC Wording and Mode Applicability	Difference/Deviation Justification
AU2	UNPLANNED loss of water level above irradiated fuel. MODE: All	AU2	UNPLANNED loss of water level above irradiated fuel MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following: (site-specific level indications). **AND** b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors. (site-specific list of area radiation monitors)	AU2.1	UNPLANNED water level drop in the REFUELING PATHWAY as indicated by level instrumentation, low water level alarm or visual observation **AND** UNPLANNED rise in corresponding area radiation levels as indicated by **any** of the following radiation monitors: • RMS-RE140 Refueling Floor Near North Entrance • RMS-RE141 Refueling Floor Near South Entrance • RMS-RE192 Fuel Building Operating Floor - South • RMS-RE193 Fuel Building Operating Floor - North	Site-specific level indications incorporated. Site-specific area radiation monitors incorporated.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
AA1	Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE. MODE: All	AA1	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	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	AA1.1	Reading on **any** Table A-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4)	The RBS radiation monitors that detect radioactivity effluent release to the environment are listed in Table A-1. UE, Alert, SAE and GE thresholds for all RBS continuously monitored gaseous and liquid release pathways are listed in Table A-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL-user.
2	Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point).	AA1.2	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.	AA1.3	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.	AA1.4	Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY: • Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min. • Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation. (Notes 1, 2)	The site boundary is the site-specific receptor point.
Notes	• The Emergency Director should declare the Alert promptly upon determining that the applicable time has been exceeded, or will likely be exceeded. • If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes. • If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes. • The pre-calculated effluent monitor values presented in EAL #1 should be used for emergency classification	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded. Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit. Note 3: If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes. Note 4 The pre-calculated effluent	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock. The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. None Incorporated site-specific EAL numbers associated with

	assessments until the results from a dose assessment using actual meteorology are available.			monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.	generic EAL#1.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
AA2	Significant lowering of water level above, or damage to, irradiated fuel. MODE: All	AA2	Significant lowering of water level above, or damage to, irradiated fuel MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Uncovery of irradiated fuel in the REFUELING PATHWAY.	AA2.1	IMMINENT uncovery of irradiated fuel in the REFUELING PATHWAY	Added the defined term "IMMINENT." Determination of irradiated fuel uncovery in the refueling pathway will always be an anticipatory determination as no direct indication is available to determine when the irradiated fuel has become uncovered.
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)	AA2.2	Damage to irradiated fuel resulting in a release of radioactivity **AND** High alarm on **any** Table A-2 radiation monitor	Site-specific list of radiation monitors are incorporated in Table A-2. Radiation monitor high alarms are specified.
3	Lowering of spent fuel pool level to (site-specific Level 2 value). [*See Developer Notes*]	AA2.3	Lowering of spent fuel pool level to 108.0 ft. (Level 2) on SFC-LI29A/B	Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (SFC-LI29A/B) capable of identifying normal level (Level 1), SFP level approximately 23 ft. above the top of the fuel racks, (Level 2) 107 ft. 10 5/16 in. (rounded to 108.0 ft. for readability) which is that level adequate to provide substantial radiation shielding for a person standing on the SFP operating deck, and SFP level at the top of the fuel racks (Level 3) 85 ft. 10 5/16 in. (rounded to 86.0 ft. for readability). RBS uses a Level 3 of approximately one foot above the highest point of any fuel rack providing

				added margin.

Table A-2 Fuel Damage Radiation Monitors
• RMS-RE140 Refueling Floor Near North Entrance • RMS-RE141 Refueling Floor Near South Entrance • RMS-RE192 Fuel Building Operating Floor - South • RMS-RE193 Fuel Building Operating Floor - North • RMS-RE5A(B) Fuel Building Ventilation Exhaust

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
AA3	Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown MODE: All	AA3	Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown MODE: All (AA3.2 Mode 3 **only**)	EAL AA3.2 mode applicability has been limited to the applicable mode of Table A-3.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Dose rate greater than 15 mR/hr in **ANY** of the following areas: • Control Room • Central Alarm Station • (other site-specific areas/rooms)	AA3.1	Dose rate > 15 mR/hr in **EITHER** of the following areas: • Control Room (RMS-RE170) • Central Alarm Station (by survey)	No other site-specific areas requiring continuous occupancy exist at RBS. The Control Room is monitored for excessive radiation by RMS-RE-170. There are no permanently installed area radiation monitors in CAS that may be used to assess this EAL threshold. Therefore, this threshold is evaluated using local radiation survey for this area.
2	An UNPLANNED event results in radiation levels that prohibit or impede access to any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode applicability identified)	AA3.2	An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table A-3 room or area (Note 5)	The site-specific list of plant rooms or areas with entry-related mode applicability are tabularized in Table A-3. The bulleted bases item "the action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections)" was removed from the list of exceptions to classification in the basis information. These actions are a consideration when the site-specific list was developed. Rooms requiring entry for these types of actions are already excluded from the list when it was developed.
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 room or area was already inoperable or out-of-service before the event occurred,	None

	no emergency classification is warranted.		then **no** emergency classification is warranted.	

Table A-3 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode**
Auxiliary Building 70' RHR B Pump Room	3
Auxiliary Building 80' RHR A Pump Room	3
Auxiliary Building 114' West	3
Control Building 95' Div 1 RSS Room	3

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
AS1	Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE MODE: All	AS1	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	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	AS1.1	Reading on **any** Table A-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4)	The RBS radiation monitors that detect radioactivity effluent release to the environment are listed in Table A-1. UE, Alert, SAE and GE thresholds for all RBS continuously monitored gaseous and liquid release pathways are listed in Table A-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL-user.
2	Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond (site-specific dose receptor point)	AS1.2	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	AS1.3	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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded. Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit. Note 3: If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes. Note 4 The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 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 RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock. The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. None Incorporated site-specific EAL numbers associated with generic EAL#1.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
AS2	Spent fuel pool level at (site-specific Level 3 description) MODE: All	AS2	Spent fuel pool level at the top of the fuel racks	Top of the fuel racks is the site-specific Level 3 description.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Lowering of spent fuel pool level to (site-specific Level 3 value)	AS2.1	Lowering of spent fuel pool level to 86.0 ft. (Level 3) on SFC-LI29A/B	For RBS, Level 3, which corresponds to approximately one foot above the highest point of any fuel rack, providing added margin, is an indicated level of: 85 ft. 10 5/16 in. (rounded to 86.0 ft. for readability) on the specified instrument.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
AG1	Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE. MODE: All	AG1	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	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values)	AG1.1	Reading on **any** Table A-1 effluent radiation monitor > column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4)	The RBS radiation monitors that detect radioactivity effluent release to the environment are listed in Table A-1. UE, Alert, SAE and GE thresholds for all RBS continuously monitored gaseous and liquid release pathways are listed in Table A-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL-user.
2	Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond (site-specific dose receptor point).	AG1.2	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 CDE greater than 5,000 mrem for	AG1.3	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.	The site boundary is the site-specific receptor point.

	one hour of inhalation.		(Notes 1, 2)	
Notes	• The Emergency Director should declare the Site Area Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded. • If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes. • If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes. • The pre-calculated effluent monitor values presented in EAL #1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.		Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.
			Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording.
			Note 3: If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.	None
			Note 4 The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual	Incorporated site-specific EAL numbers associated with generic EAL#1.

			meteorology are available.	

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
AG2	Spent fuel pool level cannot be restored to at least (site-specific Level 3 description) for 60 minutes or longer MODE: All	AG2	Spent fuel pool level **cannot** be restored to at least the top of the fuel racks for 60 minutes or longer MODE: All	Top of the fuel racks is the site-specific Level 3 description.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Spent fuel pool level cannot be restored to at least (site-specific Level 3 value) for 60 minutes or longer	AG2.1	Spent fuel pool level **cannot** be restored to at least 86.0 ft. (Level 3) on SFC-LI29A/B for ≥ 60 min. (Note 1)	For RBS, Level 3, which corresponds to approximately one foot above the highest point of any fuel rack, providing added margin, is an indicated level of: 85 ft. 10 5/16 in. (rounded to 86.0 ft. for readability) on the specified instrument.
Note	The Emergency Director should declare the General Emergency promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

Category C

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
CU1	UNPLANNED loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory for 15 minutes or longer. MODE: Cold Shutdown, Refueling	CU1	UNPLANNED loss of RPV inventory MODE: 4 - Cold Shutdown, 5 - Refueling	Deleted the words "...for 15 minutes or longer" as the 15 minute criteria only applies to EAL #1

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	UNPLANNED loss of reactor coolant results in (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level less than a required lower limit for 15 minutes or longer.	CU1.1	UNPLANNED loss of primary coolant results in RPV water level less than a required lower limit for ≥ 15 min. (Note 1)	None
2	a. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level cannot be monitored. **AND** b. UNPLANNED increase in (site-specific sump and/or tank) levels.	CU1.2	RPV water level **cannot** be monitored **AND EITHER** • UNPLANNED rise in **any** Table C-1 sump or pool level due to a loss of RPV inventory • Visual observation of UNISOLABLE RCS leakage	Added the words "...due to loss of RPV inventory" to be consistent with the IC wording. Replaced the term "increase" with the word "rise" consistent with allowed usage. Site-specific applicable sumps and tanks are listed in Table C-1 to improve the readability of the EAL. The word Tank has been replaced with Pool. There are no tanks applicable to RBS but have included the Suppression Pool as a volume where RCS leakage may relocate. Added bulleted criteria "Visual observation..." to include direct observation of significant unisolable RCS leakage.
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 the time limit has been exceeded, or will likely	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the

				be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	classification timeliness clock.

Table C-1 Sumps/Pool
• Drywell equipment drain sump • Drywell floor drain sump • Pedestal floor drain sump • CTMT equipment drain sump • CTMT floor drain sump • Suppression Pool • RHR A, B, C, HPCS, LPCS, RCIC room sumps • Auxiliary Building floor drain sump

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
CU2	Loss of all but one AC power source to emergency buses for 15 minutes or longer. MODE: Cold Shutdown, Refueling, Defueled	CU2	Loss of **all but one** AC power source to ENS buses for 15 minutes or longer. MODE: 4 - Cold Shutdown, 5 - Refueling, DEF - Defueled	"ENS buses" is the RBS-specific terminology for "emergency buses".

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. AC power capability to (site-specific emergency buses) is reduced to a single power source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS.	CU2.1	AC power capability, Table C-3, to DIV I and DIV II 4.16 KV ENS buses 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	DIV I and DIV II 4.16 KV ENS buses are the site-specific emergency buses. Site-specific AC power sources are tabularized in Table C-3.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded." to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

Table C-3 AC Power Sources
Offsite • 1RTX-XSR1C • 1RTX-XSR1D **Onsite** • EGS-EG1A • EGS-EG1B

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
CU3	UNPLANNED increase in RCS temperature MODE: Cold Shutdown, Refueling	CU3	UNPLANNED rise in RCS temperature MODE: 4 - Cold Shutdown, 5 - Refueling	Replaced the term "increase" with the word "rise" consistent with allowed usage.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit)	CU3.1	UNPLANNED rise in RCS temperature to > 200°F due to loss of decay heat removal capability	Replaced the term "increase" with the word "rise" consistent with allowed usage. 200°F is the site-specific Tech. Spec. cold shutdown temperature limit. Added "due to loss of decay heat removal capability" to reinforce the generic bases that states "EAL #1 involves a loss of decay heat removal capability."
2	Loss of **ALL** RCS temperature and (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level indication for 15 minutes or longer.	CU3.2	Loss of **all** RCS temperature and RPV water level indication for ≥ 15 min. (Note 1)	None
Note	The Emergency Director should declare the Unusual Event	N/A	Note 1: The Emergency Director should declare the event	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the

	promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded		promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
CU4	Loss of Vital DC power for 15 minutes or longer. MODE: Cold Shutdown, Refueling	CU4	Loss of Vital DC power for 15 minutes or longer. MODE 4 - Cold Shutdown, 5 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Indicated voltage is less than (site-specific bus voltage value) on required Vital DC buses for 15 minutes or longer.	CU4.1	Indicated voltage is < 105 VDC on required Safety Related DIV I and DIV II 125 VDC buses for ≥ 15 min. (Note 1)	105 VDC is the site-specific minimum vital DC bus voltage. Safety-related DC bus operability requirements are specified in Technical Specifications.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
CU5	Loss of all onsite or offsite communications capabilities. MODE: Cold Shutdown, Refueling, Defueled	CU5	Loss of **all** onsite or offsite communications capabilities. MODE: 4 - Cold Shutdown, 5 - Refueling, DEF - Defueled	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Loss of **ALL** of the following onsite communication methods: (site specific list of communications methods)	CU5.1	Loss of **all** Table C-5 onsite communication methods **OR** Loss of **all** Table C-5 State and local agency communication methods **OR** Loss of **all** Table C-5 NRC communication methods	Example EALs #1, 2 and 3 have been combined into a single EAL for simplification of presentation. Replaced "ORO" with "State and local agency" for clarification. Table C-5 provides a site-specific list of onsite, State and Local (ORO) and NRC communications methods.
2	Loss of **ALL** of the following ORO communications methods: (site specific list of communications methods)			
3	Loss of **ALL** of the following NRC communications methods: (site specific list of communications methods)			

Table C-5 Communication Methods			
System	**Onsite**	**State/Local**	**NRC**
Plant radio system	X		
Plant Paging System	X		
Sound powered phones	X		
In-plant telephones	X		
Emergency Notification System (ENS)			X
Commercial Telephone System		X	X
Satellite Phones		X	X
State of Louisiana Radio		X	
State and Local Hotline radio		X	
INFORM Notification System		X	

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
CA1	Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory MODE: Cold Shutdown, Refueling	CA1	Significant loss of RPV inventory MODE: 4 - Cold Shutdown, 5 - Refueling	Added the word "Significant…" to differentiate the Alert loss of RPV inventory IC from the Unusual Event IC which is "Unplanned loss of RPV inventory."

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory as indicated by level less than (site-specific level).	CA1.1	Loss of RPV inventory as indicated by RPV water level < -43 in. (Level 2)	-43 in. is the Level 2 actuation setpoint for HPCS and RCIC.
2	a. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level cannot be monitored for 15 minutes or longer **AND** b. UNPLANNED increase in (site-specific sump and/or tank) levels due to a loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory.	CA1.2	RPV water level **cannot** be monitored for ≥ 15 min. (Note 1) **AND EITHER** • UNPLANNED rise in **any** Table C-1 sump or pool levels due to a loss of RPV inventory • Visual observation of UNISOLABLE RCS leakage	Site-specific applicable sumps and tanks are listed in Table C-1 to improve the readability of the EAL. The word Tank has been replaced with Pool. There are no tanks applicable to RBS but have included the Suppression Pool as a volume where RCS leakage may relocate. Replaced the term "increase" with the word "rise" consistent with allowed usage. Added bulleted criteria "Visual observation" to include direct observation of significant RCS leakage.
Note	The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

			not allowed an additional 15 minutes to declare after the time limit is exceeded.	

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
CA2	Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer MODE: Cold Shutdown, Refueling, Defueled	CA2	Loss of **all** offsite and **all** onsite AC power to ENS buses for 15 minutes or longer. MODE: 4 - Cold Shutdown, 5 - Refueling, DEF - Defueled	“ENS buses” is the RBS-specific terminology for “emergency buses”.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Loss of **ALL** offsite and **ALL** onsite AC Power to (site-specific emergency buses) for 15 minutes or longer.	CA2.1	Loss of **all** offsite and **all** onsite AC power capability to DIV I and DIV II 4.16 KV ENS buses for ≥ 15 min. (Note 1)	DIV I and DIV II 4.16 KV ENS buses are the site-specific emergency buses.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added “The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded” to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
CA3	Inability to maintain the plant in cold shutdown. MODE: Cold Shutdown, Refueling	CA3	Inability to maintain plant in cold shutdown. MODE: 4 - Cold Shutdown, 5 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit) for greater than the duration specified in the following table.	CA3.1	UNPLANNED rise in RCS temperature to > 200°F for > Table C-4 duration (Note 1) **OR** UNPLANNED RPV pressure rise > 10 psig	Example EALs #1 and #2 have been combined into a single EAL as EAL #2 is the alternative threshold based on a loss of RCS temperature indication. Replaced the term "increase" with the word "rise" consistent with allowed usage. 200°F is the site-specific Tech. Spec. cold shutdown temperature limit. Table C-4 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

			time limit is exceeded.	

Table: RCS Heat-up Duration Thresholds		
RCS Status	**Containment Closure Status**	**Heat-up Duration**
Intact (but not at reduced inventory [*PWR*])	Not applicable	60 minutes*
Not intact (or at reduced inventory [*PWR*])	Established	20 minutes*
	Not Established	0 minutes
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

Table C-4 RCS Heat-up Duration Thresholds		
RCS Status	**CONTAINMENT CLOSURE Status**	**Heat-up Duration**
Intact	N/A	60 min.*
Not intact	Established	20 min.*
	Not established	0 min.
* If a 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	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
CA6	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. MODE: Cold Shutdown, Refueling	CA6	Hazardous event affecting SAFETY SYSTEMS needed for the current operating mode. MODE: 4 - Cold Shutdown, 5 - Refueling	Pluralized safety systems to be consistent with NRC EP FAQ 2016-002 that specifies degraded performance or visible damage in more than one safety system train.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. The occurrence of **ANY** of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • (site-specific hazards) • Other events with similar hazard characteristics as determined by the Shift Manager **AND** b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode. **OR** 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	CA6.1	The occurrence of **any** Table C-6 hazardous event **AND** Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode **AND EITHER:** • Event damage has caused indications of degraded performance to the second train of the SAFETY SYSTEM needed for the current operating mode • Event damage has resulted in VISIBLE DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode (Notes 9, 10)	The hazardous events have been tabularized in Table C-6. CA6.1 reflects NRC FAQ 2016-002 requiring degraded performance or visible damage to more than one train of a safety system caused by the specified events. **This wording is a deviation from NEI 99-01 Revision 6 CA6 generic wording and bases but is deemed acceptable in order to ensure that an Alert is declared only when a hazardous event causes actual or potential performance issues with safety systems. This is consistent with NRC-approved EP FAQ 2016-002.** The word “a” is replaced with “the” in the FAQ wording to provide agreement with the FAQ basis information indicating that the criteria is applicable to another train of the same safety system.

N/A	N/A	N/A	Note 9: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then emergency classification is **not** warranted.	Added Note 9 consistent with the recommendation of NRC EP FAQ 2016-002.
N/A	N/A	N/A	Note 10: If the hazardous event **only** resulted in VISIBLE DAMAGE, with **no** indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is **not** warranted.	Added Note 10 consistent with the recommendation of NRC EP FAQ 2016-002.

Table C-6 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
CS1	Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory affecting core decay heat removal capability. MODE: Cold Shutdown, Refueling	CS1	Loss of RPV inventory affecting core decay heat removal capability MODE: 4 - Cold Shutdown, 5 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. CONTAINMENT CLOSURE not established. **AND** b. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level less than (site-specific level).	CS1.1	CONTAINMENT CLOSURE **not** established **AND** RPV water level < -143 in. (Level 1)	-143 in. is the low-low-low ECCS actuation setpoint (Level 1).
2	a. CONTAINMENT CLOSURE established. **AND** b.(Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level less than (site-specific level).	CS1.2	CONTAINMENT CLOSURE established **AND** RPV water level < -162 in. (TAF)	-162 in. is the indicated RPV water level corresponding to the top of active fuel.
3	a. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level cannot be monitored for 30 minutes or longer. **AND** b. Core uncovery is indicated by **ANY** of the following:	CS1.3	RPV level **cannot** be monitored for ≥ 30 min. (Note 1) **AND** Core uncovery is indicated by **any** of the following: • UNPLANNED rise in **any** Table C-1 sump or pool levels of sufficient	Replaced the term "increase" with the word "rise" consistent with allowed usage. Site-specific applicable sumps and tanks are listed in Table C-1 to improve the readability of the EAL. The word Tank has been replaced with Pool. There are no tanks applicable to RBS but have included the Suppression Pool as a volume where RCS leakage may relocate.

	• (Site-specific radiation monitor) reading greater than (site-specific value) • Erratic source range monitor indication [*PWR*] • UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery • (Other site-specific indications)		magnitude to indicate core uncovery • Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery • RMS-RE140 Refueling Floor Near North Entrance, RMS-RE141 Refueling Floor Near South Entrance or RMS-RE16 A/B Primary containment - PAM A/B reading > 9 R/hr	Added bulleted criteria "Visual observation…" to include direct observation of significant unisolable RCS leakage The dose rate due to core shine when the top of the core becomes uncovered should result in the indicated value.
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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
CG1	Loss of (reactor vessel/RCS [*PWR*] or RPV [*BWR*]) inventory affecting fuel clad integrity with containment challenged MODE: Cold Shutdown, Refueling	CG1	Loss of RPV inventory affecting fuel clad integrity with Containment challenged MODE: 4 - Cold Shutdown, 5 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level less than (site-specific level) for 30 minutes or longer. **AND** b. **ANY** indication from the Containment Challenge Table (see below).	CG1.1	RPV level < -162 in. (TAF) for ≥ 30 min. (Note 1) **AND** **Any** Containment Challenge indication, Table C-2	-162 in. is the indicated RPV water level corresponding to the top of active fuel. 4% hydrogen concentration in the presence of oxygen is the minimum necessary to support a hydrogen burn (4%). The Igniter System is designed to prevent hydrogen accumulation by locally burning hydrogen in a controlled manner as soon as the hydrogen enters the containment atmosphere and reaches the igniters. For high rates of hydrogen production, ignition occurs at the lowest concentration that can support ignition. Following ignition, hydrogen is consumed through formation of diffusion flames where the gas enters the containment, thus controlling hydrogen concentration at approximately 4%. The MAX SAFE Operating Radiation Levels are the highest value of these parameters at which neither: (1) equipment necessary for the safe shutdown of the plant will fail, nor (2) personnel access necessary for the safe shutdown of the plant will be precluded. These are the site-specific secondary containment radiation monitor readings and are listed in EOP-3 Table SC-2. Table C-2 lists the values for equipment that is in service in Cold Shutdown.
2	a. (Reactor vessel/RCS [*PWR*] or RPV [*BWR*]) level cannot be monitored for 30 minutes	CG1.2	RCS level **cannot** be monitored for ≥ 30 min. (Note 1)	Site-specific applicable sumps and tanks are listed in Table C-1 to improve the readability of the EAL. The word Tank has been replaced with Pool. There are no tanks applicable to RBS but

	or longer. **AND** b. Core uncovery is indicated by **ANY** of the following: • (Site-specific radiation monitor) reading greater than (site-specific value) • Erratic source range monitor indication *[PWR]* • UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncovery • (Other site-specific indications) **AND** c. **ANY** indication from the Containment Challenge Table (see below).		**AND** Core uncovery is indicated by **any** of the following: • UNPLANNED rise in **any** Table C-1 sump or pool levels of sufficient magnitude to indicate core uncovery • Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery • RMS-RE140 Refueling Floor Near North Entrance, RMS-RE141 Refueling Floor Near South Entrance or RMS-RE16 A/B Primary containment - PAM A/B reading > 9 R/hr **AND** **Any** Containment Challenge indication, Table C-2	have included the Suppression Pool as a volume where RCS leakage may relocate. 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..." The dose rate due to core shine when the top of the core becomes uncovered should result in the indicated value. 4% hydrogen concentration in the presence of oxygen is the minimum necessary to support a hydrogen burn (4%). The Igniter System is designed to prevent hydrogen accumulation by locally burning hydrogen in a controlled manner as soon as the hydrogen enters the containment atmosphere and reaches the igniters. For high rates of hydrogen production, ignition occurs at the lowest concentration that can support ignition. Following ignition, hydrogen is consumed through formation of diffusion flames where the gas enters the containment, thus controlling hydrogen concentration at approximately 4%. The MAX SAFE Operating Radiation Levels are the highest value of these parameters at which neither: (1) equipment necessary for the safe shutdown of the plant will fail, nor (2) personnel access necessary for the safe shutdown of the plant will be precluded. These are the site-specific secondary containment radiation monitor readings and are listed in EOP-3 Table SC-2.
Note	The Emergency Director should declare the General Emergency promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded. N/A	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded." to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

			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 6 implements the asterisked note associated with the Containment Closure requirement.

Containment Challenge Table
■ CONTAINMENT CLOSURE not established*
■ (Explosive mixture) exists inside containment
■ UNPLANNED increase in containment pressure
■ Secondary containment radiation monitor reading above (site-specific value) [*BWR*]

* If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

Table C-2 Containment Challenge Indications

- CONTAINMENT CLOSURE **not** established (Note 6)
- Drywell or containment hydrogen concentration > 4%
- UNPLANNED rise in containment pressure
- Exceeding one or more Secondary Containment Control MAX SAFE area radiation levels:

Area	DRMS Grid 2	Max. Safe Operating Value
RHR Equip Rm A	1213	9.5E+03 mR/hr
RHR Equip Rm B	1214	9.5E+03 mR/hr
RHR Equip Rm C	1215	9.5E+03 mR/hr

Category D

Permanently Defueled Station Malfunction

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
PD-AU1 PD-AU2 PD-SU1 PD-HU1 PD-HU2 PD-HU3 PD-AA1 PD-AA2 PD-HA1 PD-HA3	Recognition Category D Permanently Defueled Station	N/A	N/A	NEI Recognition Category PD ICs and EALs are applicable only to permanently defueled stations. RBS is not a defueled station.

Category E

Independent Spent Fuel Storage Installation (ISFSI)

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
E-HU1	Damage to a loaded cask CONFINEMENT BOUNDARY MODE: All	EU1	Damage to a loaded cask CONFINEMENT BOUNDARY MODE: All	None.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than (2 times the site-specific cask specific technical specification allowable radiation level) on the surface of the spent fuel cask.	EU1.1	Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask (HI-STORM overpack) > **EITHER** of the following: • 60 mRem/hr ($\gamma + \eta$) on the top of the overpack • 600 mRem/hr ($\gamma + \eta$) on side of the overpack (excluding inlet and outlet ducts)	The specified dose rates represent 2 times the site-specific cask technical specification allowable levels per the ISFSI Technical Specifications.

Category F

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
FA1	Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier. MODE: Power Operation, Hot Standby, Startup, Hot Shutdown	FA1	**Any** loss or **any** potential loss of either Fuel Clad or RCS MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier.	FA1.1	**Any** loss or **any** potential loss of either Fuel Clad or RCS barrier (Table F-1)	Table F-1 provides the fission product barrier loss and potential loss thresholds.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
FS1	Loss or Potential Loss of any two barriers MODE: Power Operation, Hot Standby, Startup, Hot Shutdown	FS1	Loss or potential loss of **any** two barriers MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Loss or Potential Loss of any two barriers	FS1.1	Loss or potential loss of **any** two barriers (Table F-1)	Table F-1 provides the fission product barrier loss and potential loss thresholds.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
FG1	Loss of any two barriers and Loss or Potential Loss of third barrier MODE: Power Operation, Hot Standby, Startup, Hot Shutdown	FG1	Loss of **any** two barriers and loss or potential loss of the third barrier MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Loss of any two barriers and Loss or Potential Loss of third barrier	FG1.1	Loss of **any** two barriers **AND** Loss or potential loss of the third barrier (Table F-1)	Table F-1 provides the fission product barrier loss and potential loss thresholds.

BWR Fuel Clad Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI Threshold Wording	RBS FPB #(s)	RBS FPB Wording	Difference/Deviation Justification
FC Loss 1	**RCS Activity** A. (Site-specific indications that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131).	FC Loss FCB4	Coolant activity > 300 µCi/gm I-131 Dose Equivalent	300 µCi/gm DEI-131 is the site-specific indication for this reactor coolant activity.
FC Loss 2	**RPV Water Level** A. Primary containment flooding required.	FC Loss FCB1	SAP entry is required	Revised to read "SAP entry is required." Requirements for Primary Containment Flooding correspond to entry into the Severe Accident Guidelines (SAGs) and are established in EOP RPV Control, EOP RPV Control, ATWS and EOP RPV Flooding. Per the developer's guide "the phrase, "Primary containment flooding required," should be modified to agree with the site-specific EOP phrase indicating exit from all EOPs and entry to the SAGs (e.g., drywell flooding required, etc.)." Implements EP FAQ 2015-004.
FC Loss 3	**Not Applicable** Not Applicable	N/A	N/A	N/A
FC Loss 4	**Primary Containment Radiation** A. Primary containment radiation monitor reading greater than (site-specific value).	FC Loss FCB3	Containment radiation (RMS-RE16) > 3,000 R/hr	A 3,000 R/hr reading in the containment is used to indicate a loss of the Fuel Clad barrier and a release of reactor coolant, with elevated activity (300 µCi/gm dose equivalent I-131) indicative of fuel damage, into the drywell or containment.
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 RBS.

NEI FPB#	NEI Threshold Wording	RBS FPB #(s)	RBS FPB Wording	Difference/Deviation Justification
FC Loss 6	**Emergency Director Judgment** A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	FC Loss FCB5	**Any** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad Barrier	None
FC P-Loss 1	**RCS Activity** Not Applicable	N/A	N/A	N/A
FC P-Loss 2	**RPV Water Level** A. RPV water level cannot be restored and maintained above (site-specific RPV water level corresponding to the top of active fuel) or cannot be determined.	FC P-Loss FCB2	RPV water level **cannot** be restored and maintained > 162 in. (TAF) or **cannot** be determined.	-162 in. is the site-specific indicated RPV water level corresponding to the top of active fuel.
FC P-Loss 3	**Not Applicable** Not Applicable	N/A	N/A	N/A
FC P-Loss 4	**Primary Containment Radiation** Not Applicable	N/A	N/A	N/A
FC P-Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific Fuel Clad Potential Loss indication has been identified for RBS.

NEI FPB#	NEI Threshold Wording	RBS FPB #(s)	RBS FPB Wording	Difference/Deviation Justification
FC P-Loss 6	**Emergency Director Judgment** A. Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	FC P-Loss FCB6	**Any** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad Barrier	None

BWR RCS Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI IC Wording	RBS FPB #(s)	RBS FPB Wording	Difference/Deviation Justification
RCS Loss 1	**Primary Containment Pressure** A. Primary containment pressure greater than (site-specific value) due to RCS leakage.	RCS Loss RCB5	Drywell pressure > 1.68 psid due to RCS leakage	1.68 psid is the site-specific primary containment pressure corresponding to the drywell high pressure scram and isolation setpoint.
RCS Loss 2	**RPV Water Level** A. RPV water level cannot be restored and maintained above (site-specific RPV water level corresponding to the top of active fuel) or cannot be determined.	RCS Loss RCB1	RPV water level **cannot** be restored and maintained > -162 in. (TAF) or **cannot** be determined.	-162 in. is the site-specific indicated RPV water level corresponding to the top of active fuel.
RCS Loss 3	**RCS Leak Rate** A. UNISOLABLE break in ANY of the following: (site-specific systems with potential for high-energy line breaks) OR B. Emergency RPV Depressurization.	RCS Loss RCB2	UNISOLABLE break in **any** of the following: • Main Steam Line • RCIC Steam Line • RWCU • Feedwater	Main Steam Line, RCIC Steam Line, RWCU, and Feedwater are the site-specific systems with potential for high energy line breaks. Added "Failure to isolate the leak (from the Control Room or locally), within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification" to the basis. This provides agreement with the definition of "Unisolable" and ensures isolation attempts, both locally and remotely, are achieved in a timely manner.
		RCS Loss RCB3	Emergency Depressurization is required	The term "RPV" has been deleted to agree with the use of this phrase in RBS EOP-1 RPV Control "Emergency Depressurization".

NEI FPB#	NEI IC Wording	RBS FPB #(s)	RBS FPB Wording	Difference/Deviation Justification
RCS Loss 4	**Primary Containment Radiation** A. Primary containment radiation monitor reading greater than (site-specific value).	RCS Loss RCB6	Drywell radiation (RMS-RE20) > 30 R/hr	The drywell radiation monitor reading (38 R/h rounded to 30 R/hr for readability) corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals Technical Specification allowable limits. Thermally Induced Currents (TIC) discussion is added to the NEI basis information to prevent potential incorrect classification during the time period post-LOCA when the radiation monitor may be impacted by this effect.
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 RBS.
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 RCB7	**Any** condition in the opinion of the Emergency Director that indicates loss of the RCS Barrier	None
RCS P-Loss 1	**Primary Containment Pressure** Not Applicable	N/A	N/A	N/A
RCS P-Loss 2	**RPV Water Level** Not Applicable	N/A	N/A	N/A

NEI FPB#	NEI IC Wording	RBS FPB #(s)	RBS FPB Wording	Difference/Deviation Justification
RCS P-Loss 3	**RCS Leak Rate** A. UNISOLABLE primary system leakage that results in exceeding EITHER of the following: 1. Max Normal Operating Temperature OR 2. Max Normal Operating Area Radiation Level.	RCS P-Loss RCB4	UNISOLABLE primary system leakage that results in exceeding **EITHER**: • One or more EOP-3 Max Normal area radiation operating value (Table F-2) • One or more Isolation Temperature alarms (Table F-2)	Reference to EOP-3 has been added for clarification. Table F-2 was added to provide the Isolation Temperature alarm setpoints and Max Normal Operating Values from EOP-3. The alarms for the high area ambient temperature are associated with the TRM 3.3.6.1 allowable values for primary containment isolation. The use of the TRM value provides a readily identifiable condition of the RCS barrier status with an alarm in the Control Room and an isolation signal that (if the condition persists) indicates unisolable leakage as defined in the EAL threshold. The use of isolation temperatures still allows for a discernable margin prior to reaching the Max Safe Operating temperature This is an acceptable deviation from the NEI 99-01 Revision 6 guidance. Added "Failure to isolate the leak (from the Control Room or locally), within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification" to the basis. This provides agreement with the definition of "Unisolable" and ensures isolation attempts, both locally and remotely, are achieved in a timely manner.
RCS P-Loss 4	**Primary Containment Radiation** Not Applicable	N/A	N/A	N/A
RCS P-Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific RCS Potential Loss indication has been identified for RBS.
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 RCB8	**Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS Barrier	None

Table F-2 Secondary Containment Operating Values		
Area Temperatures		
Parameter	Isolation Temperature	Max Safe
Main Steam Line Tunnel	173ºF (P601-19A-A1/A3/B1/B3)	200ºF
RHR Equipment Area 1 (A)	117ºF (P601-20A-B4)	200ºF
RHR Equipment Area 2 (B)	117ºF (P601-20A-B4)	200ºF
RCIC Equipment Area	182ºF (P601-21A-B6)	200ºF
RWCU Pump Room 1 (A) / 2 (B)	165ºF (P680-1A-A2/B2)	200ºF
Area Radiation Levels		
Parameter	Max Normal	Max Safe
HPCS Area (1212) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RHR Equipment Room A (1213) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RHR Equipment Room B (1214) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RHR Equipment Room C (1215) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
LPCS Equipment Room (1216) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
HPCS Penetration Area (1217) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
LPCS Penetration Area (1218) Grid 2	8.20E+01 mR/hr	9.5E+03 mR/hr
RCIC Equipment Room (1219) Grid 2	1.20E+02 mR/hr	9.5E+03 mR/hr

BWR Containment Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI IC Wording	RBS FPB #(s)	RBS FPB Wording	Difference/Deviation Justification
PC Loss 1	**Primary Containment Conditions** A. UNPLANNED rapid drop in primary containment pressure following primary containment pressure rise OR B. Primary containment pressure response not consistent with LOCA conditions.	PC Loss CNB3	UNPLANNED rapid drop in containment pressure following containment pressure rise	The NEI phrase "primary containment" has been changed to "containment" to use terminology specific to the Mark III containment design.
		PC Loss CNB4	Containment pressure response **not** consistent with LOCA conditions	The NEI phrase "Primary containment" has been changed to "Containment" to use terminology specific to the Mark III containment design.
PC Loss 2	**RPV Water Level** Not Applicable	N/A	N/A	N/A
PC Loss 3	**Primary Containment Isolation Failure** A. UNISOLABLE direct downstream pathway to the environment exists after primary containment isolation signal OR B. Intentional primary containment venting per EOPs	PC Loss CNB9	UNISOLABLE direct downstream pathway to the environment exists after Containment isolation signal	The NEI phrase "Primary Containment" has been changed to "Containment" to use terminology specific to the Mark III containment design. Added "Failure to isolate the leak (from the Control Room or locally), within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification" to the basis. This provides agreement with the definition of "Unisolable" and ensures isolation attempts, both locally and remotely, are achieved in a timely manner.

NEI FPB#	NEI IC Wording	RBS FPB #(s)	RBS FPB Wording	Difference/Deviation Justification
	OR C. UNISOLABLE primary system leakage that results in exceeding EITHER of the following: 1. Max Safe Operating Temperature. OR 2. Max Safe Operating Area Radiation Level.	PC Loss CNB10	Intentional Containment venting per EOPs	The NEI phrase "Primary Containment" has been changed to "Containment" to use terminology specific to the Mark III containment design.
		PC Loss CNB2	UNISOLABLE primary system leakage that results in exceeding **EITHER**: • One or more EOP-3 Max Safe area radiation operating value (Table F-2) • One or more EOP-3 Max Safe area temperature operating value (Table F-2)	Reference to EOP-3 has been added for clarification. Table F-2 was added to provide the Max Safe Operating Values from EOP-3. Added "Failure to isolate the leak (from the Control Room or locally), within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification" to the basis. This provides agreement with the definition of "Unisolable" and ensures isolation attempts, both locally and remotely, are achieved in a timely manner.
PC Loss 4	**Primary Containment Radiation** Not Applicable	N/A	N/A	N/A
PC Loss 5	**Other Indications** A. (site-specific as applicable)	N/A	N/A	No other site-specific Containment Loss indication has been identified for RBS.
PC Loss 6	**Emergency Director Judgment** **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	PC Loss CNB11	**Any** condition in the opinion of the Emergency Director that indicates loss of the Containment Barrier	None

NEI FPB#	NEI IC Wording	RBS FPB #(s)	RBS FPB Wording	Difference/Deviation Justification
PC P-Loss 1	**Primary Containment Conditions** A. Primary containment pressure greater than (site-specific value) OR B. (site-specific explosive mixture) exists inside primary containment OR C. HCTL exceeded.	PC P-Loss CNB5	Containment pressure > 15 psig	15 psig is the RBS containment design pressure.
		PC P-Loss CNB6	Drywell or containment hydrogen concentration > 4%	4% hydrogen concentration is the minimum necessary to support a hydrogen burn. The Igniter System is designed to prevent hydrogen accumulation by locally burning hydrogen in a controlled manner as soon as the hydrogen enters the containment atmosphere and reaches the igniters. For high rates of hydrogen production, ignition occurs at the lowest concentration that can support ignition. Following ignition, hydrogen is consumed through formation of diffusion flames where the gas enters the containment, thus controlling hydrogen concentration at approximately 4%.
		PC P-Loss CNB7	Parameters **cannot** be restored and maintained within the safe zone of the HCTL curve (EOP Figure 2)	The NEI phrase "HCTL exceeded" has been changed to "Parameters **cannot** be restored and maintained within the safe zone of the HCTL curve (EOP Figure 2)" to use terminology consistent with the EP use of the HCTL. EOP Figure 2 is the RBS HCTL curve.
PC P-Loss 2	**RPV Water Level** A. Primary containment flooding required.	PC P-Loss CNB1	SAP entry is required	Revised to read "SAP entry is required." Requirements for Primary Containment Flooding correspond to entry into the Severe Accident Guidelines (SAGs) and are established in EOP RPV Control, EOP RPV Control - ATWS and EOP RPV Flooding. Per the developer's guide the phrase, "Primary containment flooding required," was modified to agree with the site-specific EOP phrase indicating exit from all EOPs and entry to the SAGs This difference implements EP FAQ 2015-004.

NEI FPB#	NEI IC Wording	RBS FPB #(s)	RBS FPB Wording	Difference/Deviation Justification
PC P-Loss 3	**Primary Containment Isolation Failure** Not Applicable	N/A	N/A	N/A
PC P-Loss 4	**Primary Containment Radiation** A. Primary containment radiation monitor reading greater than (site-specific value).	PC P-Loss CNB8	Containment radiation (RMS-RE16) > 12,000 R/hr	A 12,000 R/hr reading in the containment is used to indicate a potential loss of the containment barrier and a release of reactor coolant, with significant activity indicative of 20% fuel damage, into the drywell or containment. This value assumes an instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration associated with 20% clad damage into the drywell or containment atmosphere.
PC 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 RBS.
PC P-Loss 6	**Emergency Director Judgment** A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.	PC P-Loss CNB12	**Any** condition in the opinion of the Emergency Director that indicates potential loss of the Containment Barrier	None

Category H

Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
HU1	Confirmed SECURITY CONDITION or threat MODE: All	HU1	Confirmed SECURITY CONDITION or threat. MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the (site-specific security shift supervision).	HU1.1	A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by RBS Security Shift Supervision **OR** Notification of a credible security threat directed at the site **OR** A validated notification from the NRC providing information of an aircraft threat	Example EALs #1, 2 and 3 have been combined into a single EAL for ease of presentation and use.
2	Notification of a credible security threat directed at the site.			
3	A validated notification from the NRC providing information of an aircraft threat.			

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
HU2	Seismic event greater than OBE levels MODE: All	HU2	Seismic event greater than OBE levels MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Seismic event greater than Operating Basis Earthquake (OBE) as indicated by: (site-specific indication that a seismic event met or exceeded OBE limits)	HU2.1	Seismic event > OBE as indicated by **EITHER** of the following: • Annunciator P680-02A-C06, SEISMIC EVENT HIGH • Annunciator P680-02A-B06, SEISMIC EVENT HIGH/HIGH and amber lights illuminated on H13-P869 ERS-NBI101.	Annunciator P680-02A-C06, SEISMIC EVENT HIGH is actuated by ERS-NBS4B, Reactor Mat Seismic Switch (AB 70' EL) with a setpoint of 0.083 g Vertical Axis or 0.082 g Horizontal Axis. Annunciator P680-02A-B06, SEISMIC EVENT HIGH/HIGH is actuated by Reactor Mat Response Spectrum Recorder ERS-NBR2D (AB 70' EL) ((Any of 16 Mechanical Accelerometers) with setpoint ranges of .09 - 1.83g. The amber lights on H13-P869 ERS-NBI101 indicate 100% of OBE (Operating Basis Event) acceleration limits have been exceeded for the affected frequency.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
HU3	Hazardous event. MODE: All	HU3	Hazardous event MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	A tornado strike within the PROTECTED AREA.	HU3.1	A tornado strike within the PROTECTED AREA	None
2	Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.	HU3.2	Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component required by Technical Specifications for the current operating mode	Clarified the word "needed" with "required by Technical Specifications" consistent with the generic bases.
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 event external to the PROTECTED AREA involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)	Changed the term "offsite" to "external to the PROTECTED AREA" to address events located external to the PROTECTED AREA but not considered offsite.
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 RBS.

Note	EAL #3 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.	N/A	Note 7: This EAL does **not** apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.	This note, designated Note #7, is intended to apply to generic example EAL #4, not #3 as specified in the generic guidance.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
HU4	FIRE potentially degrading the level of safety of the plant. MODE: All	HU4	FIRE potentially degrading the level of safety of the plant MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. A FIRE is NOT extinguished within 15-minutes of **ANY** of the following FIRE detection indications: • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms or indications • Field verification of a single fire alarm **AND** b. The FIRE is located within **ANY** of the following plant rooms or areas: (site-specific list of plant rooms or areas)	HU4.1	A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1): • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms or indications • Field verification of a single fire alarm **AND** The FIRE is located within **any** Table H-1 area	Table H-1 provides a list of site-specific fire areas.
2	a. Receipt of a single fire alarm (i.e., no other indications of a FIRE). **AND** b. The FIRE is located within	HU4.2	Receipt of a single fire alarm (i.e., **no** other indications of a FIRE) **AND** The fire alarm is indicating a	Table H-1 provides a list of site-specific fire areas.

	ANY of the following plant rooms or areas: (site-specific list of plant rooms or areas) **AND** c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.		FIRE within **any** Table H-1 area **AND** The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1)	
3	A FIRE within the plant *or ISFSI [for plants with an ISFSI outside the plant Protected Area]* PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.	HU4.3	A FIRE within the PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)	RBS has an ISFSI located inside the plant Protected Area.
4	A FIRE within the plant *or ISFSI [for plants with an ISFSI outside the plant Protected Area]* PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.	HU4.4	A FIRE within the PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish	RBS has an ISFSI located inside 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

Table H-1 Fire Areas
• Reactor Building • Auxiliary Building • Fuel Building • Control Building • Standby Cooling Tower • Diesel Generator Building • Tunnels (B, D,E, F, G)

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
HU7	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO)UE MODE: All	HU7	Other conditions exist that in the judgment of the Emergency Director warrant declaration of a UE MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.	HU7.1	Other conditions exist which, in the judgment of the 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	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
HA1	HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes. MODE: All	HA1	HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the (site-specific security shift supervision).	HA1.1	A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by RBS Security Shift Supervision **OR** A validated notification from NRC of an aircraft attack threat within 30 min. of the site	Example EALs #1 and #2 have been combined into a single EAL for ease of use.
2	A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.			

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
HA5	Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown. MODE: All	HA5	Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown MODE: 3 – Hot Shutdown	Mode applicability has been limited to the mode restrictions of Table H-2, Mode 3 only.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. Release of a toxic, corrosive, asphyxiant or flammable gas into any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode applicability identified) **AND** b. Entry into the room or area is prohibited or impeded.	HA5.1	Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 room or area **AND** Entry into the room or area is prohibited or IMPEDED (Note 5)	The site-specific list of plant rooms or areas with entry-related mode applicability are tabularized in Table H-2. The bulleted bases item "the action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections)" was removed from the list of exceptions to classification in the basis information. These actions are a consideration when the site-specific list was developed. Rooms requiring entry for these types of actions are already excluded from the list when it was developed.
Note	**Note:** If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.	N/A	Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then **no** emergency classification is warranted.	None

Table H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode**
Auxiliary Building 70' RHR B Pump Room	3
Auxiliary Building 80' RHR A Pump Room	3
Auxiliary Building 114' West	3
Control Building 95' Div 1 RSS Room	3

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
HA6	Control Room evacuation resulting in transfer of plant control to alternate locations. MODE: All	HA6	Control Room evacuation resulting in transfer of plant control to alternate locations MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations).	HA6.1	An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels	None

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
HA7	Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert. MODE: All	HA7	Other conditions exist that in the judgment of the Emergency Director warrant declaration of an ALERT MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	HA7.1	Other conditions exist which, in the judgment of the 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	RBS IC#(s)	RBS IC Wording	Difference/Deviation 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	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision).	HS1.1	A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by RBS Security Shift Supervision	None

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
HS6	Inability to control a key safety function from outside the Control Room. MODE: All	HS6	Inability to control a key safety function from outside the Control Room MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refueling	Deleted defueled mode applicability. Control of the cited safety functions is not critical for a defueled reactor as there is no energy source in the RPV or RCS. **This is an acceptable deviation from the generic NEI 99-01 Revision 6 guidance.**

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations). **AND** b. Control of **ANY** of the following key safety functions is not reestablished within (site-specific number of minutes). • Reactivity control • Core cooling [*PWR*] / RPV water level [*BWR*] • RCS heat removal	HS6.1	An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels **AND** Control of **any** of the following key safety functions is **not** re-established within 15 min. (Note 1): • Reactivity (Modes 1 and 2 **only**) • RPV water level • RCS heat removal	The Mode applicability for the reactivity control safety function has been limited to Modes 1 and 2 (hot operating conditions). In the hot shutdown and cold operating modes adequate shutdown margin exists under all conditions. EP FAQ 2015-014. **This is an acceptable deviation from the generic NEI 99-01 Revision 6 guidance.**

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
HS7	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency. MODE: All	HS7	Other conditions exist that in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	HS7.1	Other conditions exist which, in the judgment of the 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	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
HG1	HOSTILE ACTION resulting in loss of physical control of the facility. MODE: All	N/A	N/A	IC HG1 and associated example EAL are not implemented in the RBS scheme. There are several other ICs that are redundant with this IC, and are better suited to ensure timely and effective emergency declarations. In addition, the development of new spent fuel pool level EALs, as a result of NRC Order EA-12-051, clarified the intended emergency classification level for spent fuel pool level events. **This is an acceptable deviation from the generic NEI 99-01 Revision 6 guidance.**

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision). **AND** b. **EITHER** of the following has occurred: 1. **ANY** of the following safety functions cannot be controlled or maintained. • Reactivity control • Core cooling [*PWR*]/RPV water level [*BWR*] • RCS heat removal	N/A	N/A	IC HG1 and associated example EAL are not implemented in the RBS scheme. There are several other ICs that are redundant with this IC, and are better suited to ensure timely and effective emergency declarations. In addition, the development of new spent fuel pool level EALs, as a result of NRC Order EA-12-051, clarified the intended emergency classification level for spent fuel pool level events. This deviation is justified because: 1. Hostile Action in the Protected Area is bounded by ICs HS1 and HS7. Hostile Action resulting in a loss of physical control is bound by EAL HG7, as well as any event that may lead to radiological releases to the public in excess of Environmental Protection Agency (EPA) Protective Action Guides (PAGs). a. If, for whatever reason, the Control Room must be evacuated, and control of safety functions (e.g., reactivity control, RPV water level, and RCS heat

	OR 2. Damage to spent fuel has occurred or is IMMINENT.			removal) cannot be reestablished, then IC HS6 would apply, as well as IC HS7 if desired by the EAL decision-maker. b. Also, as stated above, any event (including Hostile Action) that could reasonably be expected to have a release exceeding EPA PAGs would be bound by IC HG7. c. From a Hostile Action perspective, ICs HS1, HS7 and HG7 are appropriate, and therefore, make this part of HG1 redundant and unnecessary. d. From a loss of physical control perspective, ICs HS6, HS7 and HG7 are appropriate, and therefore, make this part of HG1 redundant and unnecessary. 2. Any event which causes a loss of spent fuel pool level will be bounded by ICs AA2, AS2 and AG2, regardless of whether it was based upon a Hostile Action or not, thus making this part of HG1 redundant and unnecessary. a. An event that leads to a radiological release will be bounded by ICs AU1, AA1, AS1 and AG1. Events that lead to radiological releases in excess of EPA PAGs will be bounded by EALs AG1 and HG7, thus making this part of HG1 redundant and unnecessary. ICs AA2, AS2, AG2, AS1, AG1, HS1, HS6, HS7 and HG7 have been implemented consistent with NEI 99-01 Revision 6 and thus HG1 is adequately bounded as described above. EP FAQ 2015-013 **This is an acceptable deviation from the generic NEI 99-01 Revision 6 guidance.**

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
HG7	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency MODE: All	HG7	Other conditions exist that in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	HG7.1	Other conditions exist which, in the judgment of the 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	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SU1	Loss of all offsite AC power capability to emergency buses for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU1	Loss of **all** offsite AC power capability to ENS buses for 15 minutes or longer MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown	"ENS buses" is the RBS-specific terminology for "emergency buses".

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Loss of **ALL** offsite AC power capability to (site-specific emergency buses) for 15 minutes or longer.	SU1.1	Loss of **all** offsite AC power capability, Table S-1, to DIV I and DIV II 4.16 KV ENS buses for ≥ 15 min. (Note 1)	DIV I and DIV II 4.16 KV ENS buses are the site-specific emergency buses. Site-specific AC power sources are listed in Table S-1.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

Table S-1 AC Power Sources
Offsite • 1RTX-XSR1C • 1RTX-XSR1D
Onsite • EGS-EG1A • EGS-EG1B

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SU2	UNPLANNED loss of Control Room indications for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU3	UNPLANNED loss of Control Room indications for 15 minutes or longer. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.	SU3.1	An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)	The site-specific Safety System 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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

[BWR parameter list]	*[PWR parameter list]*
Reactor Power	Reactor Power
RPV Water Level	RCS Level
RPV Pressure	RCS Pressure
Primary Containment Pressure	In-Core/Core Exit Temperature
Suppression Pool Level	Levels in at least (site-specific number) steam generators
Suppression Pool Temperature	Steam Generator Auxiliary or Emergency Feed Water Flow

Table S-2 Safety System Parameters
• Reactor power • RPV water level • RPV pressure • Containment pressure • Suppression Pool water level • Suppression Pool temperature

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SU3	Reactor coolant activity greater than Technical Specification allowable limits. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU4	RCS activity greater than Technical Specification allowable limits MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown	Replaced "Reactor coolant" with "RCS" consistent with site specific usage.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	(Site-specific radiation monitor) reading greater than (site-specific value).	SU4.1	Offgas Pretreatment radiation monitor high alarm (P601-22A-F03, OFF GAS PRE-TREAT HIGH RADIATION)	The High alarm indicates that the radioactivity present at the recombiner effluent discharge is approaching the Technical Specification 3.7.4 limit. The nominal setpoint of 1.5 times the full power process background radiation level ensures that the activity will not exceed a value corresponding to the Technical Specification LCO 3.7.4 allowable release rate.
2	Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications.	SU4.2	Coolant activity > 0.2 µCi/gm dose equivalent I-131 for > 48 hours **OR** Coolant activity > 4.0 µCi/gm dose equivalent I-131 instantaneous	These limits ensure the source term assumed in the safety analysis for the Main Steam Line Break (MSLB) outside containment is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the guidelines of 10 CFR 50.67. Consistent with Regulatory Guide 1.183, two cases are evaluated: (1) an equilibrium iodine case with an iodine concentration in the reactor coolant of 0.2 µCi/gm dose equivalent 1-131, and (2) an iodine spiking case with an iodine concentration in the reactor coolant of 4.0 µCi/gm dose equivalent I-131.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SU4	RCS leakage for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU5	RCS leakage for 15 minutes or longer MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	RCS unidentified or pressure boundary leakage greater than (site-specific value) for 15 minutes or longer.	SU5.1	RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. (Note 1) **OR** RCS identified leakage > 25 gpm for ≥ 15 min. (Note 1) **OR** Leakage from the RCS to a location outside Containment > 25 gpm for ≥ 15 min. (Note 1)	Example EALs #1, 2 and 3 have been combined into a single EAL for usability. Added “Failure to isolate the leak (from the Control Room or locally), within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification” to the basis. This provides agreement with the definition of “Unisolable” and ensures isolation attempts, both locally and remotely, are achieved in a timely manner
2	RCS identified leakage greater than (site-specific value) for 15 minutes or longer.			
3	Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer.			
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added “The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded” to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

			exceeded.	

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SU5	Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor. MODE: Power Operation	SU6	Automatic or manual scram fails to shut down the reactor MODE: 1 - Power Operation, 2 - Startup	Mode 2 – Startup has been included. For BWRs, including RBS, the plant operating mode is defined by the position of the Reactor Mode Switch. During a normal plant startup the Reactor Mode Switch is placed in the Startup position (Startup Mode 2) as reactor power is increased. Typically reactor power is increased to ~7-8% before the Reactor Mode Switch is placed in Run (Power Operations Mode 1). 5% reactor power (APRM downscale) is the site-specific indication of a successful reactor scram. Therefore it is appropriate to include Startup Mode 2 to failure to scram ICs.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. An automatic (trip [PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor.	SU6.1	An automatic scram did **not** shut down the reactor as indicated by reactor power > 5% after **any** RPS setpoint is exceeded **AND** A subsequent automatic scram or manual scram action taken at the reactor control console (Mode Switch, Manual PBs, ARI) is successful in shutting down the reactor as indicated by reactor power ≤ 5% (APRM downscale)	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. Added the words "... as indicated by reactor power > 5% after **any** RPS setpoint is exceeded" to clarify that it is a failure of the automatic trip when a valid trip signal has been exceed. Mode Switch, Manual PBs, and initiation of ARI are the manual actions taken to shut down the reactor.
2	a. A manual trip ([PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. **EITHER** of the following: 1. A subsequent manual action taken at the reactor	SU6.2	A manual scram did **not** shut down the reactor as indicated by reactor power > 5% after **any** manual trip action was initiated **AND** A subsequent automatic scram or manual scram action taken at	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. Added the words "... as indicated by reactor power > 5% after **any** manual trip action was initiated" to clarify that it is a failure of any

	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.		the reactor control console (Mode Switch, Manual PBs, ARI) is successful in shutting down the reactor as indicated by reactor power ≤ 5% (APRM downscale) (Note 8)	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. Mode Switch, Manual PBs, and initiation of ARI are the manual actions taken to shut down the reactor.
Notes	**Note:** A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.	N/A	Note 8: A manual action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.	None

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SU6	Loss of all onsite or offsite communications capabilities. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU7	Loss of **all** onsite or offsite communications capabilities. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Loss of **ALL** of the following onsite communication methods: (site-specific list of communications methods)	SU7.1	Loss of **all** Table S-4 onsite communication methods **OR** Loss of **all** Table S-4 State and local agency communication methods **OR** Loss of **all** Table S-4 NRC communication methods	Example EALs #1, 2 and 3 have been combined into a single EAL for simplification of presentation. Replaced "ORO" with "State and local agency" for clarification Table S-4 provides a site-specific list of onsite, State and Local (ORO) and NRC communications methods.
2	Loss of **ALL** of the following ORO communications methods: (site-specific list of communications methods)			
3	Loss of **ALL** of the following NRC communications methods: (site-specific list of communications methods)			

Table S-4 Communication Methods			
System	**Onsite**	**State/Local**	**NRC**
Plant radio system	X		
Plant Paging System	X		
Sound powered phones	X		
In-plant telephones	X		
Emergency Notification System (ENS)			X
Commercial Telephone System		X	X
Satellite Phones		X	X
State of Louisiana Radio		X	
State and Local Hotline radio		X	
INFORM Notification System		X	

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SU7	Failure to isolate containment or loss of containment pressure control. [*PWR*] MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	N/A	N/A	This IC and its associated example EALs are applicable to PWRs only and therefore not included.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. Failure of containment to isolate when required by an actuation signal. **AND** b. **ALL** required penetrations are not closed within 15 minutes of the actuation signal.	N/A	N/A	This IC and its associated example EALs are applicable to PWRs only and therefore not included.
2	a. Containment pressure greater than (site-specific pressure). **AND** b. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer.			
N/A	N/A	N/A	N/A	This IC and its associated example EALs are applicable to PWRs only and therefore not included.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SA1	Loss of all but one AC power source to emergency buses for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SA1	Loss of **all but one** AC power source to ENS buses for 15 minutes or longer. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown	"ENS buses" is the RBS-specific terminology for "emergency buses".

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. AC power capability to (site-specific emergency buses) is reduced to a single power source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in a loss of all AC power to SAFETY SYSTEMS.	SA1.1	AC power capability, Table S-1, to DIV I and DIV II 4.16 KV ENS buses 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	DIV I and DIV II 4.16 KV ENS buses are the site-specific emergency buses. Site-specific AC power sources are listed in Table S-1.
Note	The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

Table S-1 AC Power Sources
Offsite • 1RTX-XSR1C • 1RTX-XSR1D **Onsite** • EGS-EG1A • EGS-EG1B

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SA2	UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SA3	UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer. **AND** **ANY** of the following transient events in progress. • Automatic or manual runback greater than 25% thermal reactor power • Electrical load rejection greater than 25% full electrical load • Reactor scram [*BWR*] / trip [*PWR*] • ECCS (SI) actuation • Thermal power oscillations greater than 10% [*BWR*]	SA3.1	An UNPLANNED event results in the inability to monitor **one or more** Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1) **AND** **Any** significant transient is in progress, Table S-3	The site-specific Safety System Parameter list is in Table S-2. The NEI phrase "Primary Containment" has been changed to "Containment" to use terminology specific to the Mark III containment design. The significant transient list has been tabularized in Table S-3 for ease of use.

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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

[*BWR parameter list*]	[*PWR parameter list*]
Reactor Power	Reactor Power
RPV Water Level	RCS Level
RPV Pressure	RCS Pressure
Primary Containment Pressure	In-Core/Core Exit Temperature
Suppression Pool Level	Levels in at least (site-specific number) steam generators
Suppression Pool Temperature	Steam Generator Auxiliary or Emergency Feed Water Flow

Table S-2 Safety System Parameters
• Reactor power • RPV water level • RPV pressure • Containment pressure • Suppression Pool water level • Suppression Pool temperature

Table S-3 Significant Transients
• Reactor scram • Runback > 25% thermal reactor power • Electrical load rejection > 25% full electrical load • ECCS injection • Thermal power oscillations > 10%

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SA5	Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor. MODE: Power Operation	SA6	Automatic or manual scram fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are **not** successful in shutting down the reactor MODE: 1 - Power Operation, 2 - Startup	Mode 2 – Startup has been included. For BWRs, including RBS, the plant operating mode is defined by the position of the Reactor Mode Switch. During a normal plant startup the Reactor Mode Switch is placed in the Startup position (Startup Mode 2) as reactor power is increased. Typically reactor power is increased to ~7-8% before the Reactor Mode Switch is placed in Run (Power Operations Mode 1). 5% reactor power (APRM downscale) is the site-specific indication of a successful reactor scram. Therefore it is appropriate to include Startup Mode 2 to failure to scram ICs.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. Manual actions taken at the reactor control consoles are not successful in shutting down the reactor.	SA6.1	An automatic or manual scram fails to shut down the reactor as indicated by reactor power > 5% **AND** Manual scram actions taken at the reactor control console (Mode Switch, Manual PBs, ARI) are **not** successful in shutting down the reactor as indicated by reactor power > 5% (Note 8)	As specified in the generic developers guidance "Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level)." Reactor power ≤ 5% is the site-specific indication of a successful reactor scram. Mode Switch, Manual PBs, and initiation of ARI are the manual actions taken to shut down the reactor.
Notes	**Note:** A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.	N/A	Note 8: A manual action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or	None

				implementation of boron injection strategies	

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SA9	Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SA8	Hazardous event affecting SAFETY SYSTEMS needed for the current operating mode MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown	Pluralized safety systems to be consistent with NRC EP FAQ 2016-002 that specifies degraded performance or visible damage in more than one safety system train.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. The occurrence of **ANY** of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • (site-specific hazards) • Other events with similar hazard characteristics as determined by the Shift Manager **AND** b. **EITHER** of the following: 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode. **OR** 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	SA8.1	The occurrence of **any** Table S-5 hazardous event **AND** Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode **AND EITHER:** • Event damage has caused indications of degraded performance to the second train of the SAFETY SYSTEM needed for the current operating mode • Event damage has resulted in VISIBLE DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode (Notes 9, 10)	The hazardous events have been tabularized in Table S-5. SA8.1 reflects NRC FAQ 2016-002 requiring degraded performance or visible damage to more than one train of a safety system caused by the specified events. **This wording is a deviation from NEI 99-01 Revision 6 SA9 generic wording and bases but is deemed acceptable in order to ensure that an Alert is declared only when a hazardous event causes actual or potential performance issues with safety systems. This is consistent with NRC-approved EP FAQ 2016-002.** The word "a" is replaced with "the" in the FAQ wording to provide agreement with the FAQ basis information indicating that the criteria is applicable to another train of the same safety system.

N/A	N/A	N/A	Note 9: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then emergency classification is **not** warranted	Added Note 9 consistent with the recommendation of NRC EP FAQ 2016-002.
N/A	N/A	N/A	Note 10: If the hazardous event **only** resulted in VISIBLE DAMAGE, with **no** indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is **not** warranted.	Added Note 10 consistent with the recommendation of NRC EP FAQ 2016-002.

Table 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 Shift Manager

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SS1	Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SS1	Loss of **all** offsite and **all** onsite AC power to ENS buses for 15 minutes or longer MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown	"ENS buses" is the RBS-specific terminology for "emergency buses".

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Loss of **ALL** offsite and **ALL** onsite AC power to (site-specific emergency buses) for 15 minutes or longer.	SS1.1	Loss of **all** offsite and **all** onsite AC power capability to DIV I and DIV II 4.16 KV ENS buses for ≥ 15 min. (Note 1)	DIV I and DIV II 4.16 KV ENS buses are the site-specific emergency buses.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SS5	Inability to shutdown the reactor causing a challenge to (core cooling [*PWR*] / RPV water level [*BWR*]) or RCS heat removal. MODE: Power Operation	SS6	Inability to shut down the reactor causing a challenge to RPV water level or RCS heat removal MODE: 1 - Power Operation, 2 - Startup	Mode 2 – Startup has been included. For BWRs, including RBS, the plant operating mode is defined by the position of the Reactor Mode Switch. During a normal plant startup the Reactor Mode Switch is placed in the Startup position (Startup Mode 2) as reactor power is increased. Typically reactor power is increased to ~7-8% before the Reactor Mode Switch is placed in Run (Power Operations Mode 1). 5% reactor power (APRM downscale) is the site-specific indication of a successful reactor scram. Therefore it is appropriate to include Startup Mode 2 to failure to scram ICs.

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor. **AND** b. All manual actions to shutdown the reactor have been unsuccessful. **AND** c. **EITHER** of the following conditions exist: • (Site-specific indication of an inability to adequately remove heat from the core) • (Site-specific indication of an inability to adequately remove heat from the RCS)	SS6.1	An automatic or manual scram 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**: RPV water level **cannot** be restored and maintained > -187 in. **OR** Heat Capacity Temperature Limit (HCTL) exceeded (EOP Figure 2)	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" scram action. Indication of an inability to adequately remove heat from the core occurs when RPV water level cannot be restored and maintained above -187 in., which is the EOP RPV water level indicative of a loss of adequate core cooling. Indication of an inability to adequately remove heat from the RCS occurs when parameters cannot be restored and maintained within the safe region of the HCTL curve.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SS8	Loss of all Vital DC power for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SS2	Loss of **all** vital DC power for 15 minutes or longer. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	Indicated voltage is less than (site-specific bus voltage value) on **ALL** (site-specific Vital DC busses) for 15 minutes or longer.	SS2.1	Indicated voltage is < 105 VDC on Safety Related DIV I and DIV II 125 VDC buses for ≥ 15 min. (Note 1)	105 VDC is the site-specific minimum vital DC bus voltage. Safety Related DIV I and DIV II 125 VDC buses are the site-specific credited vital DC buses.
Note	The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added “The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded” to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SG1	Prolonged loss of all offsite and all onsite AC power to emergency buses. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SG1a	Prolonged loss of **all** offsite and **all** onsite AC power to ENS buses MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown	"ENS buses" is the RBS-specific terminology for "emergency buses".

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. Loss of **ALL** offsite and **ALL** onsite AC power to (site-specific emergency buses). **AND** b. **EITHER** of the following: • Restoration of at least one AC emergency bus in less than (site-specific hours) is not likely. • (Site-specific indication of an inability to adequately remove heat from the core)	SG1.1	Loss of **all** offsite and **all** onsite AC power capability to DIV I and DIV II 4.16 KV ENS buses **AND EITHER:** • Restoration of at least one 4.16 KV ENS bus in < 4 hours is **not** likely (Note 1) • RPV water level **cannot** be restored and maintained > -187 in.	DIV I and DIV II 4.16 KV ENS buses are the site-specific emergency buses. 4 hours is the site-specific SBO coping analysis time. Indication of an inability to adequately remove heat from the core occurs when RPV water level cannot be restored and maintained above -187 in., which is the EOP RPV water level indicative of a loss of adequate core cooling.
Note	The Emergency Director should declare the General Emergency promptly upon determining that (site-specific hours) has been exceeded, or will likely be exceeded.	N/A	Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.

			additional 15 minutes to declare after the time limit is exceeded.	

NEI IC#	NEI IC Wording	RBS IC#(s)	RBS IC Wording	Difference/Deviation Justification
SG8	Loss of all AC and Vital DC power sources for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SG1b	Loss of **all** ENS AC and vital DC power sources for 15 minutes or longer MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown	"ENS AC" is the RBS-specific terminology for "emergency AC".

NEI Ex. EAL #	NEI Example EAL Wording	RBS EAL #	RBS EAL Wording	Difference/Deviation Justification
1	a. Loss of **ALL** offsite and **ALL** onsite AC power to (site-specific emergency buses) for 15 minutes or longer. **AND** b. Indicated voltage is less than (site-specific bus voltage value) on ALL (site-specific Vital DC busses) for 15 minutes or longer.	SG1.2	Loss of **all** offsite and **all** onsite AC power capability to DIV I and DIV II 4.16 KV ENS buses for ≥ 15 min. (Note 1) **AND** Indicated voltage is < 105 VDC on Safety Related DIV I and DIV II 125 VDC buses for ≥ 15 min. (Note 1)	DIV I and DIV II 4.16 KV ENS buses are the site-specific emergency buses. 105 VDC is the site-specific minimum vital DC bus voltage. Safety Related DIV I and DIV II 125 VDC buses are the safety-related DC buses that are the credited.
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 the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.	The classification timeliness note has been standardized across the RBS EAL scheme by referencing the "time limit" specified within the EAL wording. Added "The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded" to reinforce the concept that the EAL timing component runs concurrent with the classification timeliness clock.