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10 CFR 50.90
10 CFR 50 Appendix E

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U. S. Nuclear Regulatory Commission
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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261 / RENEWED LICENSE NO. DPR-23

Proposed Changes to the Emergency Plan

In accordance with the provisions of 10 CFR 50.90 and 10 CFR 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," Section IV.B, Duke Energy is submitting a license amendment request to change the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2) Emergency Plan.

Duke Energy is proposing to change the Emergency Action Levels (EALs) from a scheme based upon Revision 4 of Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," to one based upon Revision 6 of NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors." This change in scheme requires NRC approval prior to implementation.

The enclosures to this letter provide descriptions and assessments of the proposed changes. The enclosures also provide the existing Emergency Plan pages redline and strikeout to show the proposed changes:

- Enclosure 1 - Evaluation of Proposed Change
- Enclosure 2 - Comparison Matrix of NEI 99-01, Rev. 6 Generic Guidance to Proposed HBRSEP2 Emergency Classification System
- Enclosure 3 - Emergency Action Level Technical Bases Document (Clean Version)
- Enclosure 4 - Emergency Action Level Technical Bases (Redline and Strikeout Version)
- Enclosure 5 - Supporting Calculation EAL Table R-1, "Effluent Monitor Classification Thresholds"
- Enclosure 6 - Emergency Action Level Wallcharts for HBRSEP2

Duke Energy requests approval of the proposed changes by May 31, 2016, with the amendment being implemented within 180 days of issuance.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated South Carolina State Officials.

Duke Energy commits to review the new classification scheme with state and local emergency management officials following NRC approval and prior to implementation.

Duke Energy is currently pursuing the Fukushima Orders required Spent Fuel Pool Level Instrumentation and the qualification of the equipment for use with the proposed EAL scheme; however, if the equipment will not be available at the time of implementation of the proposed EALs, Duke Energy will pursue a change to the License Amendment Request.

Please address any comments or questions regarding this matter to Mr. Richard Hightower, Manager – Nuclear Regulatory Affairs at (843) 857-1329.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 13, 2015.

Sincerely,



R. Michael Glover
Site Vice President

RMG/msc

Enclosures

1. Evaluation of Proposed Change
2. Comparison Matrix of NEI 99-01, Rev. 6 Generic Guidance to Proposed HBRSEP2 Emergency Classification System
3. Emergency Action Level Technical Bases Document (Clean Version)
4. Emergency Action Level Technical Bases (Redline and Strikeout Version)
5. Supporting Calculation EAL Table R-1, "Effluent Monitor Classification Thresholds"
6. Emergency Action Level Wallcharts for HBRSEP2

cc: Mr. V. M. McCree, NRC, Region II
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Ms. S. E. Jenkins, Manager, Infectious and Radioactive Waste Management Section (SC)

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Enclosure 1

EVALUATION OF PROPOSED CHANGE

Subject: Proposed Changes to the Emergency Plan

1.0 SUMMARY DESCRIPTION

2.0 DETAILED DESCRIPTION

3.0 TECHNICAL EVALUATION

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

4.2 Significant Hazards Consideration

4.3 Conclusions

5.0 ENVIRONMENTAL CONSIDERATION

6.0 REFERENCES

1.0 SUMMARY DESCRIPTION

In accordance with the provisions of 10 CFR 50.90 and 10 CFR 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," Section IV.B, Duke Energy is submitting a license amendment request (LAR) to change the H. B. Robinson Team Electric Plant Unit No. 2 (HBRSEP2) Emergency Plan.

The proposed changes involve upgrading selected HBRSEP2 Emergency Action Levels (EALs) based on Nuclear Energy Institute (NEI) 99-01, Revision 6, "Methodology for Development of Emergency Action Levels," using the guidance of NRC Regulatory Issue Summary (RIS) 2003-18, Supplement 2, "Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels." HBRSEP2 currently uses an emergency classification scheme based on NEI 99-01, "Methodology for Development of Emergency Action Levels," Revision 4 (endorsed by the Nuclear Regulatory Commission (NRC) in Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," Revision 5, June 2005) and approved for HBRSEP2 in Reference 6). The plan, as changed, would continue to meet the standards in 10 CFR 50.47(b) and the requirements in Appendix E to 10 CFR 50.

2.0 DETAILED DESCRIPTION

HBRSEP2 currently uses an emergency classification scheme based on NEI 99-01, "Methodology for Development of Emergency Action Levels," Revision 4 (endorsed by the Nuclear Regulatory Commission (NRC) in Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," Revision 5, June 2005). Duke Energy requests approval to change the HBRSEP2 scheme basis to that described in NEI 99-01, Revision 6.

3.0 TECHNICAL EVALUATION

The Initiating Conditions (ICs) and EALs that comprise the proposed scheme are presented in Enclosure 2. This matrix provides a cross-reference between each generic IC and EAL contained in NEI 99-01, Rev. 6 and the proposed HBRSEP2-specific IC and EAL. Differences and Deviations are identified in accordance with the guidance discussed in RIS 2003-18 (and Supplements). The basis for each Difference or Deviation is also included. No Deviations from NEI 99-01, Rev. 6 have been made.

The matrix follows the presentation order of NEI 99-01, Rev. 6 - Abnormal Rad Levels/Radiological Effluent, Cold Shutdown/Refueling System Malfunction, Events Related to Independent Spent Fuel Storage Installation (ISFSI), Fission Product Barrier Degradation, Hazards and Other Conditions Affecting Plant Safety, and System Malfunction. The Defueled Station section is not used since HBRSEP2 is an operating unit.

Differences and Deviations

As discussed in RIS 2003-18, Supplement 1, dated July 13, 2004, differences and Deviations are Defined as follows:

- 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.).

Enclosure 2 identifies each Difference between NEI 99-01 and the final products being evaluated in this LAR. These differences do not alter the meaning or intent of the ICs or EALs. There are no Deviations between NEI 99-01 and the final products being evaluated in this LAR.

Incorporation of EALFAQs

Where appropriate, information from Emergency Action Level Frequently Asked Questions (EALFAQs) has been incorporated into Enclosure 2 and Enclosure 3.

Related Documents

Enclosure 3 includes the site-specific technical basis document for each recognition category for the proposed scheme. These documents include appropriate information from the basis information contained in NEI 99-01, Revision 6. A redline and strikeout version is provided as Enclosure 4.

Enclosure 5 contains the supporting calculation for HBRSEP2 EAL Table R-1, "Effluent Monitor Classification Thresholds."

Enclosure 6 contains the revised HBRSEP2 EAL Wallcharts.

Operational Modes and Applicability

Mode applicability of the proposed ICs and EALs is consistent with the NEI 99-01, Revision 6 basis scheme. The Operating Modes for HBRSEP2, as defined in the Technical Specifications, are listed below.

| MODE | TITLE | REACTIVITY CONDITION (k_{eff}) | % RATED THERMAL POWER | Average Reactor Coolant Temperature (°F) |
|------|-----------------|--|-----------------------------|--|
| 1 | Power Operation | ≥ 0.99 | > 5 | NA |
| 2 | Startup | ≥ 0.99 | ≤ 5 | NA |
| 3 | Hot Standby | < 0.99 | NA | ≥ 350 |
| 4 | Hot Shutdown | < 0.99 | NA | $350 > T_{avg} > 20$ |
| 5 | Cold Shutdown | < 0.99 | NA | ≤ 200 |
| 6 | Refueling | NA | NA | NA |

In addition to these operating modes, NEI 99-01, Revision 6 defines the following additional Mode:

Defueled- All reactor fuel removed from Reactor Vessel (full core off load during refueling or extended outage)

Station procedures recognize, and are consistent with, this Mode definition.

State / Local Government Review of Proposed Changes

Duke Energy meets monthly with the North Carolina and South Carolina state and local emergency management agencies. The state and local emergency management officials are advised of any EAL changes actually implemented. In the case of this EAL scheme revision, Duke Energy has committed to provide a review of the new classification scheme to state and local emergency management officials following NRC approval and prior to implementation.

Implementation Description

Duke Energy plans to implement the proposed emergency classification scheme in the third quarter of 2016 (within six months of NRC approval this LAR). When implemented, the changes to the EALs presented in Enclosure 3 will become effective. The Emergency Action Level Technical Basis Documents (Enclosure 3) will be revised and maintained as a training and background reference resource. Any changes to the approved ICs and EALs will be made in accordance with 10 CFR 50.54(q).

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.47(b)(4) states, "A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures."

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

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

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

Regulatory Guide 1.101 Revision 4, Section C. Regulatory Position states, "The guidance in NUMARC/NESP-007 (Revision 2, January 1992), "Methodology for Development of Emergency Action Levels," is acceptable to the NRC staff as an alternative method to that described in Appendix 1 to NUREG-0654/FEMA-REP-1 for developing EALs required in Section IV.B of Appendix E to 10 CFR Part 50 and 10 CFR 50.47(b)(4). In addition, the guidance contained in NEI 99-01 (Revision 4, January 2003), "Methodology for Development of Emergency Action Levels," is acceptable to the NRC staff as an alternative method to that described in Appendix 1 to NUREG-0654/FEMA-REP-1 and NUMARC/NESP-007 for developing EALs required in Section IV of Appendix E to 10 CFR Part 50 and 10 CFR 50.47(b)(4)."

4.2 Significant Hazards Consideration

Duke Energy has evaluated whether or not a significant hazards consideration (SHC) is warranted with the proposed changes by addressing the three criterion set forth in 10 CFR 50.92(c) as discussed below.

Criterion 1:

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

Response: No.

These changes affect the HBRSEP2 Emergency Plan and do not alter any of the requirements of the Operating License or the Technical Specifications. The proposed changes do not modify any plant equipment and do not impact any failure modes that could lead to an accident. Additionally, the proposed changes do not impact the consequence of any analyzed accident since the changes do not affect any equipment related to accident mitigation. Based on this discussion, the proposed amendment does not increase the probability or consequences of an accident previously evaluated.

Criterion 2:

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

Response: No.

These changes affect the HBRSEP2 Emergency Plan and do not alter any of the requirements of the Operating License or the Technical Specifications. They do not modify any plant equipment and there is no impact on the capability of the existing equipment to perform their intended functions. No system setpoints are being modified and no changes are being made to the method in which plant operations are conducted. No new failure modes are

introduced by the proposed changes. The proposed amendment does not introduce an accident initiator or malfunctions that would cause a new or different kind of accident. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3:

Does the proposed amendment involve a significant reduction in a margin of safety? Response: No.

These changes affect the HBRSEP2 Emergency Plan and do not alter any of the requirements of the Operating License or the Technical Specifications. The proposed changes do not affect any of the assumptions used in the accident analysis, nor do they affect any operability requirements for equipment important to plant safety. Therefore, the proposed changes will not result in a significant reduction in the margin of safety as defined in the bases for technical specifications covered in this license amendment request.

4.3 Conclusion

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

Duke Energy has determined that the proposed amendment would not change requirements with respect to use of a facility component located within the restricted area, as defined by 10 CFR 20, nor would it change inspection or surveillance requirements. Duke Energy has evaluated the proposed change and has determined that the change does not involve:

- I. A Significant Hazards Consideration,
- II. A significant change in the types or significant increase in the amounts of any effluent that may be released off site, or
- III. A significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10)(ii). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or

environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

- (1) NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels"
- (2) NEI 99-01, Revision 6, "Methodology for Development of Emergency Action Levels"
- (3) NEI 99-01, Revision 4, "Methodology for Development of Emergency Action Levels"
- (4) NRC Regulatory Issue Summary 2003-18, Supplement 2, "Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels"
- (5) NRC Regulatory Issue Summary 2005-02, "Clarifying the Process for Making Emergency Plan Changes"
- (6) Letter from Marlayna Vaaler (USNRC) to Mr. Thomas D. Walt (Duke Energy) dated September 14, 2007, "H. B. Robinson Steam Electric Plant, Unit No. 2 - Emergency Action Level Revisions (TAC No. MD3327)" (ML072540561)

RNP-RA/15-0034
Enclosure 2
112 Pages (including cover page)

Enclosure 2

COMPARISON MATRIX OF NEI 99-01, REV. 6 GENERIC
GUIDANCE TO PROPOSED HBRSEP2 EMERGENCY
CLASSIFICATION SYSTEM



**Robinson Nuclear Plant
NEI 99-01 Revision 6
EAL Comparison Matrix**

Revision 0 [4/24/15]

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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 the Robinson Nuclear Plant (RNP) ICs, Mode Applicability and EALs. This document provides a means of assessing RNP differences and deviations from the NRC endorsed guidance given in NEI 99-01. Discussion of RNP 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.

RNP has taken no deviations from the generic guidance.

Comparison Matrix Format

The ICs and EALs discussed in this document are grouped according to NEI 99-01 Recognition Categories. Within each Recognition Category, the ICs and EALs are listed in tabular format according to the order in which they are given in NEI 99-01. Generally, each row of the comparison matrix provides the following information:

- NEI EAL/IC identifier
- NEI EAL/IC wording
- RNP EAL/IC identifier
- RNP EAL/IC wording
- Description of any differences or deviations

EAL Wording

In Section 4.1, NEI recommends the following: “The guidance in NEI 99-01 is not intended to be applied to plants “as-is”; however, developers should attempt to keep their site-specific schemes as close to the generic guidance as possible. The goal is to meet the intent of the generic Initiating Conditions (ICs) and Emergency Action Levels (EALs) within the context of site-specific characteristics – locale, plant design, operating features, terminology, etc. Meeting this goal will result in a shorter and less cumbersome NRC review and approval process, closer alignment with the schemes of other nuclear

power plant sites and better positioning to adopt future industry-wide scheme enhancements”

EAL Emphasis Techniques

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

The print and paragraph formatting conventions summarized below guide presentation of the RNP 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.

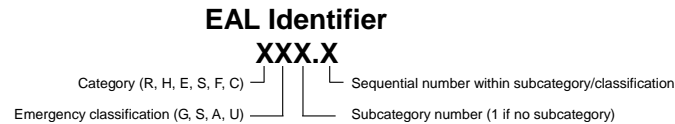
Global Differences

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 RNP EALs appear as unique EALs (e.g., HU3.1 through HU3.4).
3. Mode applicability identifiers (numbers/letter) modify the NEI 99-01 mode applicability names as follows: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown, 5 - Cold Shutdown, 6 - Refueling, D – Defueled. NEI 99-01 defines Defueled as follows: “Reactor Vessel contains no irradiated fuel (full core off-load during refueling or extended outage).”
4. NEI 99-01 uses the terms greater than, less than, greater than or equal to, etc. in the wording of some example EALs. For consistency and reduce EAL-user reading burden, RNP has adopted use of boolean symbols in place of the NEI 99-01 text modifiers within the EAL wording.
5. “min.” is the standard abbreviation for “minutes” and is used to reduce EAL user reading burden.
6. The term “Emergency Director” has been replaced by “Emergency Coordinator” consistent with site-specific nomenclature.
7. Wherever the generic bracketed PWR term “reactor vessel/RCS” is provided, RNP uses the term “RCS” as the site-specific nomenclature.
8. IC/EAL identification:
 - NEI Recognition Category A “Abnormal Radiation Levels/ Radiological Effluents” has been changed to Category R “Abnormal Rad Levels / Rad Effluents.” The designator “R” is more intuitively associated with radiation (rad) or radiological events. NEI IC designators beginning with “A” have likewise been changed to “R.”

- NEI 99-01 defines the thresholds requiring emergency classification (example EALs) and assigns them to ICs which, in turn, are grouped in “Recognition Categories.” The RNP IC/EAL scheme includes the following features:
 - a. Division of the NEI EAL set into three groups:
 - EALs applicable under all plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
 - EALs applicable only under hot operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Hot Standby, Startup or Power Operation mode.
 - EALs applicable only under cold operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling or Defueled mode.

The purpose of the groups is to avoid review of hot condition EALs when the plant is in a cold condition and avoid review of cold condition EALs when the plant is in a hot condition. This approach significantly minimizes the total number of EALs that must be reviewed by the EAL-user for a given plant condition 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 RNP EAL categories/subcategories and their relationship to NEI Recognition Categories are listed in Table 1.
 - c. Unique identification of each EAL – Four characters comprise the EAL identifier as illustrated in Figure 1.

Figure 1 – EAL Identifier

The first character is a letter associated with the category in which the EAL is located. The second character is a letter associated with the emergency classification level (G for General Emergency, S for Site Area Emergency, A for Alert, and U for Notification of Unusual Event). The third character is a number associated with one or more subcategories within a given category. Subcategories are sequentially numbered beginning with the number “1”. If a category does not have a subcategory, this character is assigned the number “1”. The fourth character is a number preceded by a period for each EAL within a subcategory. EALs are sequentially numbered within the emergency classification level of a subcategory beginning with the number “1”.

The EAL identifier is designed to fulfill the following objectives:

- Uniqueness – The EAL identifier ensures that there can be no confusion over which EAL is driving the need for emergency classification.
- Speed in locating the EAL of concern – When the EALs are displayed in a matrix format, knowledge of the EAL identifier alone can lead the EAL-user to the location of the EAL within the classification matrix. The identifier conveys the category, subcategory and classification level. This assists ERO responders (who may not be in the same facility as the ED) to find the EAL of concern in a timely manner without the need for a word description of the classification threshold.
- Possible classification upgrade – The category/subcategory/identifier scheme helps the

EAL-user find higher emergency classification EALs that may become active if plant conditions worsen.

Table 2 lists the RNP 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 RNP 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 RNP 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 RNP (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 RNP equipment/instrument identification and/or noun names to EALs.
- Combining like ICs that are exactly the same but have different operating modes as long as the intent of each IC is maintained and the overall progression of the EAL scheme is not affected.
- Any change to the IC and/or EAL, and/or basis wording, as stated in NEI 99-01, that does not alter the intent of the IC and/or EAL, i.e., the IC and/or EAL continues to:
 - Classify at the correct classification level.
 - Logically integrate with other EALs in the EAL scheme.
 - Ensure that the resulting EAL scheme is complete (i.e., classifies all potential emergency conditions).

The following are examples of deviations:

- Use of altered mode applicability.
- Altering key words or time limits.
- Changing words of physical reference (protected area, safety-related equipment, etc.).
- Eliminating an IC. This includes the removal of an IC from the Fission Product Barrier Degradation category as this impacts the logic of Fission Product Barrier ICs.
- Changing a Fission Product Barrier from a Loss to a Potential Loss or vice-versa.
- Not using NEI 99-01 definitions as the intent is for all NEI 99-01 users to have a standard set of defined terms as defined in NEI 99-01. Differences due to plant types are permissible (BWR or PWR). Verbatim compliance to the wording in NEI 99-01 is not necessary as long as the intent of the defined word is maintained. Use of the wording provided in NEI 99-01 is encouraged since the intent is for all users to have a standard set of defined terms as defined in NEI 99-01.

- Any change to the IC and/or EAL, and/or basis wording as stated in NEI 99-01 that does alter the intent of the IC and/or EAL, i.e., the IC and/or EAL:
 - Does not classify at the classification level consistent with NEI 99-01.
 - Is not logically integrated with other EALs in the EAL scheme.
 - Results in an incomplete EAL scheme (i.e., does not classify all potential emergency conditions).

The “Difference Justification” columns in the remaining sections of this document identify each difference between the NEI 99-01 IC/EAL wording and the RNP 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 effect 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. RNP has identified no deviations from the NEI 99-01 guidance as represented in Table 3.

Table 1 – RNP EAL Categories/Subcategories

| RNP EALs | | NEI Recognition Category |
|---|---|--|
| Category | Subcategory | |
| <u>Group: Any Operating Mode:</u> | | |
| R – Abnormal Rad Levels/Rad Effluent | 1 – Radiological Effluent 2 – Irradiated Fuel Event 3 – Area Radiation Levels | Abnormal 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 Gases 6 – Control Room Evacuation 7 – Emergency Coordinator Judgment | Hazards and Other Conditions Affecting Plant Safety ICs/EALs |
| E – ISFSI | N/A | ISFSI ICs/EALs |
| <u>Group: Hot Conditions:</u> | | |
| S – System Malfunction | 1 – Loss of Emergency AC Power 2 – Loss of Vital DC Power 3 – Loss of Control Room Indications 4 – RCS Activity 5 – RCS Leakage 6 – RPS Failure 7 – Loss of Communications 8 – Containment Failure 9 – Hazardous Event Affecting Safety Systems | System Malfunction ICs/EALs |
| F – Fission Product Barrier | None | Fission Product Barrier ICs/EALs |
| <u>Group: Cold Conditions:</u> | | |
| C – Cold Shutdown/Refueling System Malfunction | 1 – RCS Level 2 – Loss of Emergency AC Power 3 – RCS Temperature 4 – Loss of Vital DC Power 5 – Loss of Communications 6 – Hazardous Event Affecting Safety Systems | Cold Shutdown./ Refueling System Malfunction ICs/EALs |

Table 2 – NEI / RNP EAL Identification Cross-Reference

| NEI | | RNP | |
|-----|-------------|---|-------|
| IC | Example EAL | Category and Subcategory | EAL |
| AU1 | 1 | R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent | RU1.1 |
| AU1 | 2 | R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent | RU1.1 |
| AU1 | 3 | R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent | RU1.2 |
| AU2 | 1 | R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event | RU2.1 |
| AA1 | 1 | R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent | RA1.1 |
| AA1 | 2 | R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent | RA1.2 |
| AA1 | 3 | R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent | RA1.3 |
| AA1 | 4 | R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent | RA1.4 |
| AA2 | 1 | R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event | RA2.1 |
| AA2 | 2 | R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event | RA2.2 |
| AA2 | 3 | R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event | RA2.3 |
| AA3 | 1 | R – Abnormal Rad Levels / Rad Effluent, 3 – Area Radiation Levels | RA3.1 |
| AA3 | 2 | R – Abnormal Rad Levels / Rad Effluent, 3 – Area Radiation Levels | RA3.2 |
| AS1 | 1 | R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent | RS1.1 |
| AS1 | 2 | R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent | RS1.2 |
| AS1 | 3 | R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent | RS1.3 |

| NEI | | RNP | |
|-----|-------------|---|-------|
| IC | Example EAL | Category and Subcategory | EAL |
| AS2 | 1 | R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event | RS2.1 |
| AG1 | 1 | R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent | RG1.1 |
| AG1 | 2 | R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent | RG1.2 |
| AG1 | 3 | R – Abnormal Rad Levels / Rad Effluent, 1 – Radiological Effluent | RG1.3 |
| AG2 | 1 | R – Abnormal Rad Levels / Rad Effluent, 2 – Irradiated Fuel Event | RG2.1 |
| CU1 | 1 | C – Cold SD/ Refueling System Malfunction, 1 – RCS Level | CU1.1 |
| CU1 | 2 | C – Cold SD/ Refueling System Malfunction, 1 – RCS Level | CU1.2 |
| CU2 | 1 | C – Cold SD/ Refueling System Malfunction, 2 – Loss of 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 – RCS Level | CA1.1 |
| CA1 | 2 | C – Cold SD/ Refueling System Malfunction, 1 – RCS Level | CA1.2 |
| CA2 | 1 | C – Cold SD/ Refueling System Malfunction, 1 – Loss of 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 – RCS Level | CS1.1 |

| NEI | | RNP | |
|-------|-------------|--|-------|
| IC | Example EAL | Category and Subcategory | EAL |
| CS1 | 2 | C – Cold SD/ Refueling System Malfunction, 1 – RCS Level | CS1.2 |
| CS1 | 3 | C – Cold SD/ Refueling System Malfunction, 1 – RCS Level | CS1.3 |
| CG1 | 1 | C – Cold SD/ Refueling System Malfunction, 1 – RCS Level | CG1.1 |
| CG1 | 2 | C – Cold SD/ Refueling System Malfunction, 1 – RCS Level | CG1.2 |
| E-HU1 | 1 | E – ISFSI | EU1.1 |
| FA1 | 1 | F – Fission Product Barrier Degradation | FA1.1 |
| FS1 | 1 | F – Fission Product Barrier Degradation | FS1.1 |
| FG1 | 1 | F – Fission Product Barrier Degradation | FG1.1 |
| HU1 | 1, 2, 3 | H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security | HU1.1 |
| HU2 | 1 | H – Hazards and Other Conditions Affecting Plant Safety, 2 – Seismic Event | HU2.1 |
| HU3 | 1 | H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technological Hazard | HU3.1 |
| HU3 | 2 | H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technological Hazard | HU3.2 |
| HU3 | 3 | H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technological Hazard | HU3.3 |
| HU3 | 4 | H – Hazards and Other Conditions Affecting Plant Safety, 3 – Natural or Technological Hazard | HU3.4 |
| HU3 | 5 | N/A | N/A |
| HU4 | 1 | H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire | HU4.1 |
| HU4 | 2 | H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire | HU4.2 |
| HU4 | 3 | H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire | HU4.3 |

| NEI | | RNP | |
|-----|-------------|--|-------|
| IC | Example EAL | Category and Subcategory | EAL |
| HU4 | 4 | H – Hazards and Other Conditions Affecting Plant Safety, 4 – Fire | HU4.4 |
| HU7 | 1 | H – Hazards and Other Conditions Affecting Plant Safety, 7 – EC Judgment | HU7.1 |
| HA1 | 1, 2 | H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security | HA1.1 |
| HA5 | 1 | H – Hazards and Other Conditions Affecting Plant Safety, 5 – Hazardous Gases | HA5.1 |
| HA6 | 1 | H – Hazards and Other Conditions Affecting Plant Safety, 6 – Control Room Evacuation | HA6.1 |
| HA7 | 1 | H – Hazards and Other Conditions Affecting Plant Safety, 7 – EC Judgment | HA7.1 |
| HS1 | 1 | H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security | HS1.1 |
| HS6 | 1 | H – Hazards and Other Conditions Affecting Plant Safety, 6 – Control Room Evacuation | HS6.1 |
| HS7 | 1 | H – Hazards and Other Conditions Affecting Plant Safety, 7 – EC Judgment | HS7.1 |
| HG1 | 1 | H – Hazards and Other Conditions Affecting Plant Safety, 1 – Security | HG1.1 |
| HG7 | 1 | H – Hazards and Other Conditions Affecting Plant Safety, 7 – EC 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 | SU4.2 |
| SU3 | 2 | S – System Malfunction, 4 – RCS Activity | SU4.1 |
| 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 | | RNP | |
|-----|-------------|--|-------|
| IC | Example EAL | Category and Subcategory | EAL |
| SU6 | 1, 2, 3 | S – System Malfunction, 7 – Loss of Communications | SU7.1 |
| SU7 | 1, 2 | S – System Malfunction, 8 – Containment Failure | SU8.1 |
| SA1 | 1 | S – System Malfunction, 1 – Loss of Emergency AC Power | SA1.1 |
| SA2 | 1 | S – System Malfunction, 3 – Loss of Control Room Indications | SA3.1 |
| SA5 | 1 | S – System Malfunction, 6 – RPS Failure | SA6.1 |
| SA9 | 1 | S – System Malfunction, 9 – Hazardous Event Affecting Safety Systems | SA9.1 |
| SS1 | 1 | S – System Malfunction, 1 – Loss of Emergency AC Power | SS1.1 |
| SS5 | 1 | S – System Malfunction, 6 – RPS Failure | SS6.1 |
| SS8 | 1 | S – System Malfunction, 2 – Loss of Vital DC Power | SS2.1 |
| SG1 | 1 | S – System Malfunction, 1 – Loss of Emergency AC Power | SG1.1 |
| SG8 | 1 | S – System Malfunction, 1 – Loss of Emergency AC Power | SG1.2 |

Table 3 – Summary of Deviations

| NEI | | RNP EAL | Description |
|-----|-------------|------------|-------------|
| IC | Example EAL | | |
| N/A | N/A | N/A | N/A |

Category A

Abnormal Rad Levels / Radiological Effluent

| NEI IC# | NEI IC Wording and Mode Applicability | RNP IC#(s) | RNP IC Wording and Mode Applicability | Difference Justification |
|---------|---|------------|---|--|
| AU1 | Release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer. MODE: All | RU1 | Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer MODE: All | The RNP ODCM is the site-specific effluent release controlling document. |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|---|
| 1 | Reading on ANY effluent radiation monitor greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer: (site-specific monitor list and threshold values corresponding to 2 times the controlling document limits) | RU1.1 | Reading on any Table R-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3) | <p>Example EALs #1 and #2 have been combined into a single EAL.</p> <p>The NEI phrase "...effluent radiation monitor greater than 2 times the (site-specific effluent release controlling document)" and "effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit " have been replaced with "...any Table R-1 effluent radiation monitor > column "UE".</p> <p>UE thresholds for all RNP continuously monitored gaseous release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL user. The values shown in Table R-1 column "UE", consistent with the NEI bases, represent two times the ODCM release limits for both liquid and gaseous release.</p> |
| 2 | Reading on ANY effluent radiation monitor greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer. | | | |
| 3 | Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the (site-specific effluent release controlling document) limits for | RU1.2 | Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2) | The RNP ODCM is the site-specific effluent release controlling document. |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|--|---|
| | 60 minutes or longer. | | | |
| Notes | <ul style="list-style-type: none"> 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 | <p>Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.</p> <p>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.</p> <p>Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.</p> | <p>The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording.</p> <p>The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording.</p> <p>None</p> |

| Table R-1 Effluent Monitor Classification Thresholds | | | | | | |
|--|-----------------------|------------|-------------|---------------|---------------|-------------|
| Release Point | | Monitor | GE | SAE | Alert | UE |
| Gaseous | Plant Vent | R-14C | --- | --- | --- | 2.16E+5 cpm |
| | | R-14D | 6.38E+5 cpm | 6.38E+4 cpm | 6.38E+3 cpm | --- |
| | | R-14E | 3.31E+3 cpm | 3.40E+2 cpm | 4.30E+1 cpm | --- |
| | FHB Exhaust | R-20 | --- | --- | --- | 8.06E+5 cpm |
| | FHB Exhaust HR | R-30 | --- | 2.60E+4 mR/hr | 2.60E+3 mR/hr | --- |
| Liquid | Liquid Waste Disposal | R-18 | ---- | ---- | ---- | 4.08E+6 cpm |
| | SGBD Effluent | R-19 A/B/C | ---- | ---- | ---- | 6.94E+5 cpm |
| | Condensate Polisher | R-37 | ---- | ---- | ---- | 4.23E+5 cpm |

| NEI IC# | NEI IC Wording and Mode Applicability | RNP IC#(s) | RNP IC Wording and Mode Applicability | Difference Justification |
|---------|---|------------|--|--------------------------|
| AU2 | UNPLANNED loss of water level above irradiated fuel. MODE: All | RU2 | Unplanned loss of water level above irradiated fuel MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|--|--|
| 1 | <p>a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by ANY of the following: (site-specific level indications).</p> <p>AND</p> <p>b. UNPLANNED rise in area radiation levels as indicated by ANY of the following radiation monitors. (site-specific list of area radiation monitors)</p> | RU2.1 | <p>UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication</p> <p>AND</p> <p>UNPLANNED rise in corresponding area radiation levels as indicated by any of the following:</p> <ul style="list-style-type: none"> • R-2 CV Area • R-5 Spent Fuel Pit Area • Local area survey | <p>The site-specific list of radiation monitors are listed.</p> <p>Added the word “corresponding” to emphasize the cause and effect intent of the EAL.</p> |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| AA1 | Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE. MODE: All | RA1 | Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|---|--|
| 1 | Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values) | RA1.1 | Reading on any Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4) | The RNP radiation monitors that detect radioactivity effluent release to the environment are listed in Table R-1. UE, Alert, SAE and GE thresholds for all RNP continuously monitored gaseous and liquid release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL-user. |
| 2 | Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point). | RA1.2 | Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 3, 4) | The site boundary is the site-specific receptor point. |
| 3 | Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond (site-specific dose receptor point) for one hour of exposure. | RA1.3 | Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure (Notes 1, 2) | The site boundary is the site-specific receptor point. |

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| 4 | <p>Field survey results indicate EITHER of the following at or beyond (site-specific dose receptor point):</p> <ul style="list-style-type: none"> ● Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer. ● Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation. | RA1.4 | <p>Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY:</p> <ul style="list-style-type: none"> ● 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. <p>(Notes 1, 2)</p> | The site boundary is the site-specific receptor point. |
| Notes | <ul style="list-style-type: none"> ● 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 | <p>Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.</p> <p>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.</p> <p>Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.</p> <p>Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification</p> | <p>The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording.</p> <p>The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording.</p> <p>None</p> <p>Incorporated site-specific EAL numbers associated with generic EAL#1.</p> |

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| | assessments until the results from a dose assessment using actual meteorology are available. | | assessments until the results from a dose assessment using actual meteorology are available. | |
|--|--|--|--|--|

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|---|--------------------------|
| AA2 | Significant lowering of water level above, or damage to, irradiated fuel. MODE: All | RA2 | Significant lowering of water level above, or damage to, irradiated fuel MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|--|---|
| 1 | Uncovery of irradiated fuel in the REFUELING PATHWAY. | RA2.1 | Uncovery of irradiated fuel in the REFUELING PATHWAY | None |
| 2 | Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by ANY of the following radiation monitors: (site-specific listing of radiation monitors, and the associated readings, setpoints and/or alarms) | RA2.2 | Damage to irradiated fuel resulting in a release of radioactivity AND A high alarm on any of the following: <ul style="list-style-type: none"> • R-2 CV Area • R-5 Spent Fuel Pit Area • R-11/R-12 Process Monitor CV Air and Plant Vent (when sampling CV) • R-14 Plant Vent • R-21 Fuel Handling Building Upper Level | The NEI phrase "...from the fuel as indicated by ANY of the following radiation monitors" has been replaced with "...AND A high alarm on any of the following" for clarification that the classification requires two conditions: damage to fuel and a resultant high radiation alarm. The site-specific list of radiation monitors are listed in bullet format for ease of reading. The high setpoints for the radiation monitors are indicative of significant increases in area and/or airborne radiation. |
| 3 | Lowering of spent fuel pool level to (site-specific Level 2 value). [See <i>Developer Notes</i>] | RA2.3 | Lowering of spent fuel pool level to \leq 24 ft. | Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3). The SFP level instruments consist of a primary channel (LI-11442A & LI-11443A) and back-up channel (LI-11442B & LI-11443B) each spanning approximately 24 ft. (14 ft. – 38 ft. indicated). Level 2 corresponds to an indicated SFP level of |

| | | | | |
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| | | | | 24 ft. or approximately 10 ft. above the top of the SFP racks |
|--|--|--|--|---|

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|---|--------------------------|
| AA3 | Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown MODE: All | RA3 | Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|--|
| 1 | Dose rate greater than 15 mR/hr in ANY of the following areas: <ul style="list-style-type: none"> Control Room Central Alarm Station (other site-specific areas/rooms) | RA3.1 | Dose rate > 15 mR/hr in EITHER of the following areas: <p>Control Room (R-1)</p> <p>OR</p> <p>Central Alarm Station (by survey)</p> | No other site-specific areas requiring continuous occupancy exist at RNP. R-1 monitors the Control room for area radiation. The CAS does not have installed area radiation monitoring and thus must be determined by survey. |
| 2 | An UNPLANNED event results in radiation levels that prohibit or impede access to any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode applicability identified) | RA3.2 | An UNPLANNED event results in radiation levels that prohibit or impede access to any Table R-2/H-2 rooms or areas (Note 5) | Table R-2/H-2 contains the site-specific list of plant rooms or areas with entry-related mode applicability identified. |
| Note | If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is | 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 | None |

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| | warranted. | | warranted. | |
|--|------------|--|------------|--|

| Table R-2/H-2 Safe Operation & Shutdown Rooms/Areas | |
|---|----------------|
| Room/Area | Mode(s) |
| Reactor Auxiliary Building, 1 st level hallway | 1,2,3,4,5 |
| Reactor Auxiliary Building, 2 nd level hallway | 1,2,3,4,5 |
| Charging Pump Room | 1,2,3,4,5 |
| Component Cooling Water Pump Room | 1,2,3,4,5 |
| Primary Sample Room | 1,2,3,4,5 |
| Primary Demineralizer Room | 1,2,3 |
| Spent Fuel Pump / Heat Exchanger Room | 1,2,3,4,5 |
| Pipe Alley | 4 |
| RHR Heat Exchanger Room | 4 |
| RHR Pump Room entry area (access to RHR Pump CCW flow indication / control) | 4 |
| Boric Acid Batch Tank Room | 1,2,3,4,5 |
| Emergency Bus E1/E2 Room | 3,4,5 |
| Turbine Building 1 st Floor (includes Condensate Polisher, Makeup Water Treatment and Secondary Sample Room) | 1,2,3,4 |
| Turbine Building 2 nd Floor | 1,2,3,4 |
| Turbine Building 3 rd Floor | 1,3,4 |
| Containment Building | 3 |

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

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|--|
| 1 | Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values) | RS1.1 | Reading on any Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4) | The RNP radiation monitors that detect radioactivity effluent release to the environment are listed in Table R-1. UE, Alert, SAE and GE thresholds for all RNP continuously monitored gaseous and liquid release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL-user. |
| 2 | Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond (site-specific dose receptor point) | RS1.2 | Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 3, 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): <ul style="list-style-type: none"> Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer. Analyses of field survey samples indicate thyroid | RS1.3 | Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY: <ul style="list-style-type: none"> 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 | The site boundary is the site-specific receptor point. |

| | | | | |
|-------|--|--|--|--|
| | CDE greater than 500 mrem for one hour of inhalation. | | inhalation. (Notes 1, 2) | |
| Notes | <ul style="list-style-type: none"> ● 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. | | <p>Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.</p> <p>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.</p> <p>Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.</p> <p>Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.</p> | <p>The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording.</p> <p>The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording.</p> <p>None</p> <p>Incorporated site-specific EAL numbers associated with generic EAL#1.</p> |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|---|
| AS2 | Spent fuel pool level at (site-specific Level 3 description) MODE: All | RS2 | Spent fuel pool level at the top of the fuel racks | Top of the fuel racks is the site-specific Level 3. |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|--|
| 1 | Lowering of spent fuel pool level to (site-specific Level 3 value) | RS2.1 | Lowering of spent fuel pool level to ≤ 14.75 ft. | <p>Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3).</p> <p>The SFP level instruments consist of a primary channel (LI-11442A & LI-11443A) and back-up channel (LI-11442B & LI-11443B) each spanning approximately 24 ft. (14 ft. – 38 ft. indicated). Level 3 (top of the spent fuel racks) corresponds to an SFP level of 14 ft. However, the level instruments can actually only measure to 14.75 ft.</p> |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| AG1 | Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE. MODE: All | RG1 | Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|--|---|
| 1 | Reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site-specific monitor list and threshold values) | RG1.1 | Reading on any Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4) | The RNP radiation monitors that detect radioactivity effluent release to the environment are listed in Table R-1. UE, Alert, SAE and GE thresholds for all RNP continuously monitored gaseous or liquid release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL-user. |
| 2 | Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond (site-specific dose receptor point). | RG1.2 | Dose assessment using actual meteorology indicates doses > 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 3, 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): <ul style="list-style-type: none"> ● Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer. ● Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for | RG1.3 | Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY: <ul style="list-style-type: none"> ● Closed window dose rates > 1,000 mR/hr expected to continue for ≥ 60 min. ● Analyses of field survey samples indicate thyroid CDE > 5,000 mrem for 60 min. of inhalation. | The site boundary is the site-specific receptor point. |

| | | | | |
|-------|--|--|--|--|
| | one hour of inhalation. | | (Notes 1, 2) | |
| Notes | <ul style="list-style-type: none"> ● 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. | | <p>Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.</p> <p>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.</p> <p>Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.</p> <p>Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.</p> | <p>The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording.</p> <p>The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording.</p> <p>None</p> <p>Incorporated site-specific EAL numbers associated with generic EAL#1.</p> |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|---|---|
| AG2 | Spent fuel pool level cannot be restored to at least (site-specific Level 3 description) for 60 minutes or longer MODE: All | RG2 | Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer | Top of the fuel racks is the site-specific Level 3. |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|---|
| 1 | Spent fuel pool level cannot be restored to at least (site-specific Level 3 value) for 60 minutes or longer | RG2.1 | Spent fuel pool level cannot be restored to at least 14.75 ft. for ≥ 60 min. (Note 1) | Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3). The SFP level instruments consist of a primary channel (LI-11442A & LI-11443A) and back-up channel (LI-11442B & LI-11443B) each spanning approximately 24 ft. (14 ft. – 38 ft. indicated). Level 3 (top of the spent fuel racks) corresponds to an SFP level of 14 ft. However, the level instruments can actually only measure to 14.75 ft. |
| 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 Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

Category C

Cold Shutdown / Refueling System Malfunction

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| CU1 | UNPLANNED loss of (reactor vessel/RCS [<i>PWR</i>] or RCP [<i>BWR</i>]) inventory for 15 minutes or longer. MODE: Cold Shutdown, Refueling | CU1 | UNPLANNED loss of RCS inventory for 15 minutes or longer MODE: 5 - Cold Shutdown, 6 - Refueling | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|--|--|
| 1 | UNPLANNED loss of reactor coolant results in (reactor vessel/RCS [<i>PWR</i>] or RCP [<i>BWR</i>]) level less than a required lower limit for 15 minutes or longer. | CU1.1 | UNPLANNED loss of reactor coolant results in RCS water level less than a required lower limit for ≥ 15 min. (Note 1) | None |
| 2 | a. (Reactor vessel/RCS [<i>PWR</i>] or RCP [<i>BWR</i>]) level cannot be monitored. AND b. UNPLANNED increase in (site-specific sump and/or tank) levels. | CU1.2 | RCS water level cannot be monitored AND EITHER <ul style="list-style-type: none"> UNPLANNED increase in any Table C-1 sump or tank due to a loss of RCS inventory Visual observation of UNISOLABLE RCS leakage | Added the phrase “due to a loss of RCS inventory” because the NEI basis states: “Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.” Table C-1 lists site-specific applicable sumps and tanks. Added bulleted criteria “Visual observation of UNISOLABLE RCS leakage” to include direct observation of RCS leakage. |
| Note | The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded. | N/A | Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

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|--|--|--|---------------------------------------|--|
| | | | exceeded, or will likely be exceeded. | |
|--|--|--|---------------------------------------|--|

| Table C-1 Sumps / Tanks |
|---|
| <ul style="list-style-type: none">• Containment (CV) sump• PRT• RCDT• CCW surge tank |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|---|--------------------------|
| CU2 | Loss of all but one AC power source to emergency buses for 15 minutes or longer. MODE: Cold Shutdown, Refueling, Defueled | CU2 | Loss of all but one AC power source to emergency buses for 15 minutes or longer. MODE: 6 - Cold Shutdown, 6 - Refueling, D - Defueled | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|--|--|
| 1 | a. AC power capability to (site-specific emergency buses) is reduced to a single power source for 15 minutes or longer. AND b. Any additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS. | CU2.1 | AC power capability to 480V emergency buses E-1 and E-2 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 | 480V buses E-1 and E-2 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 Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

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

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|---|--|
| 1 | UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit) | CU3.1 | UNPLANNED increase in RCS temperature to > 200°F due to loss of decay heat removal capability | 200°F is the site-specific Tech. Spec. cold shutdown temperature limit. Added "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 [<i>PWR</i>] or RCP [<i>BWR</i>]) level indication for 15 minutes or longer. | CU3.2 | Loss of all RCS temperature and RCS level indication for ≥ 15 min. (Note 1) | None |
| Note | The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded | N/A | Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|--|--------------------------|
| CU4 | Loss of Vital DC power for 15 minutes or longer. MODE: Cold Shutdown, Refueling | CU4 | Loss of Vital DC power for 15 minutes or longer. MODE: 5 - Cold Shutdown, 6 - Refueling | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|--|
| 1 | Indicated voltage is less than (site-specific bus voltage value) on required Vital DC buses for 15 minutes or longer. | CU4.1 | < 109.5 VDC (Bus A) / < 106.2 (Bus B) bus voltage indications on Technical Specification required 125 VDC buses for ≥ 15 min. (Note 1) | The specified VDC values are the site-specific calculated minimum vital DC bus voltages. DC operability requirements are specified in Technical Specifications. |
| Note | The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded. | N/A | Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|---|--------------------------|
| CU5 | Loss of all onsite or offsite communications capabilities. MODE: Cold Shutdown, Refueling, Defueled | CU5 | Loss of all onsite or offsite communications capabilities. MODE: Cold Shutdown, 6 - Refueling, D - Defueled | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|---|--|
| 1 | Loss of ALL of the following onsite communication methods: (site specific list of communications methods) | CU5.1 | Loss of all Table C-4 onsite communication methods OR Loss of all Table C-4 offsite communication methods OR Loss of all Table C-4 NRC communication methods | Example EALs #1, 2 and 3 have been combined into a single EAL. Table C-4 provides a site-specific list of onsite, offsite and NRC communications methods. Replaced the acronym "ORO" with the word "offsite". ORO is not a standard acronym. |
| 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-4 Communication Methods | | | |
|--|---------------|------------|------------|
| System | Onsite | ORO | NRC |
| Public Address System | X | | |
| PBX Telephone System | X | | |
| Radio Transceivers for RNP and Vicinity | X | | |
| Back-up Telephone System (ESSX) | X | | |
| Plant Security Radio Transceivers | X | | |
| Corporate Telephone Communications System (Voicenet) | | X | X |
| BellSouth | | X | X |
| Dedicated Telephone System to Load Dispatcher | | X | |
| Plant Security Radio Control Station | | X | |
| DEMNET | | X | |
| NRC Emergency Telecommunication System (ETS) | | | X |
| Satellite Phones | | X | X |
| Cellular Phones | | X | X |
| Palmetto 800 Transceivers | | X | |

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

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|---|--|
| 1 | Loss of (reactor vessel/RCS [PWR] or RCP [BWR]) inventory as indicated by level less than (site-specific level). | CA1.1 | Loss of RCS inventory as indicated by RCS water level < - 72 in. (69% RVLIS Full Range) | -72 in. (69% RVLIS Full Range) RCS level indication corresponds to the level requiring RHR pumps to be tripped. |
| 2 | a. (Reactor vessel/RCS [PWR] or RCP [BWR]) level cannot be monitored for 15 minutes or longer AND b. UNPLANNED increase in (site-specific sump and/or tank) levels due to a loss of (reactor vessel/RCS [PWR] or RCP [BWR]) inventory. | CA1.2 | RCS water level cannot be monitored for ≥ 15 min. (Note 1) AND EITHER <ul style="list-style-type: none"> • UNPLANNED increase in any Table C-1 sump or tank due to a loss of RCS inventory • Visual observation of UNISOLABLE RCS leakage | Added the phrase “due to a loss of RCS inventory” because the NEI basis states: “Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.” Table C-1 lists site-specific applicable sumps and tanks. Added bulleted criteria “Visual observation of UNISOLABLE RCS leakage” to include direct observation of RCS leakage. |
| Note | The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded | N/A | Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| CA2 | Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer MODE: Cold Shutdown, Refueling, Defueled | CA2 | Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer. MODE: Cold Shutdown, 6 - Refueling, D - Defueled | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|--|
| 1 | Loss of ALL offsite and ALL onsite AC Power to (site-specific emergency buses) for 15 minutes or longer. | CA2.1 | Loss of all offsite and all onsite AC power capability to 480V emergency buses E-1 and E-2 for ≥ 15 min. (Note 1) | 480V buses E-1 and E-2 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 Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|---|--------------------------|
| CA3 | Inability to maintain the plant in cold shutdown. MODE: Cold Shutdown, Refueling | CA3 | Inability to maintain the plant in cold shutdown. MODE: Cold Shutdown, 6 - Refueling | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|--|---|
| 1 | UNPLANNED increase in RCS temperature to greater than (site-specific Technical Specification cold shutdown temperature limit) for greater than the duration specified in the following table. | CA3.1 | UNPLANNED increase in RCS temperature to > 200°F for > Table C-3 duration (Note 1) OR UNPLANNED RCS pressure increase > 10 psig due to a loss of RCS cooling (this does not apply during water-solid plant conditions) | Example EALs #1 and #2 have been combined into a single EAL as EAL # is the alternative threshold based on a loss of RCS temperature indication. 200°F is the site-specific Tech. Spec. cold shutdown temperature limit. Table C-3 is the site-specific implementation of the generic RCS Heat-up Duration Threshold table. 10 psig is the site-specific pressure increase readable by Control Room indications. |
| 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 Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

| Table: RCS Heat-up Duration Thresholds | | |
|---|----------------------------|------------------|
| RCS Status | Containment Closure Status | Heat-up Duration |
| Intact (but not at reduced inventory [<i>PWR</i>]) | Not applicable | 60 minutes* |
| Not intact (or at reduced inventory [<i>PWR</i>]) | Established | 20 minutes* |
| | Not Established | 0 minutes |
| * If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable. | | |

| Table C-3: RCS Heat-up Duration Thresholds | | |
|--|----------------------------|------------------|
| RCS Status | Containment Closure Status | Heat-up Duration |
| Intact (but not REDUCED INVENTORY) | N/A | 60 min.* |
| Not intact OR At REDUCED INVENTORY | established | 20 min.* |
| | not established | 0 min. |
| * If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable. | | |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|--|--------------------------|
| CA6 | Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. MODE: Cold Shutdown, Refueling | CA6 | Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. MODE: Cold Shutdown, 6 - Refueling | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|------------------|---|--------------|--|--|
| 1 | <p>a. The occurrence of ANY of the following hazardous events:</p> <ul style="list-style-type: none"> ● 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 <p>AND</p> <p>b. EITHER of the following:</p> <ol style="list-style-type: none"> 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode. <p>OR</p> <ol style="list-style-type: none"> 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode. | CA6.1 | <p>The occurrence of any Table C-5 hazardous event</p> <p>AND EITHER:</p> <ul style="list-style-type: none"> ● Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode ● The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode | The hazardous events have been tabularized in Table C-5. |

| Table C-5 Hazardous Events |
|--|
| <ul style="list-style-type: none">● 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 | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| CS1 | Loss of (reactor vessel/RCS [<i>PWR</i>] or RCP [<i>BWR</i>]) inventory affecting core decay heat removal capability. MODE: Cold Shutdown, Refueling | CS1 | Loss of RCS inventory affecting core decay heat removal capability MODE: Cold Shutdown, 6 - Refueling | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|---|
| 1 | a. CONTAINMENT CLOSURE not established. AND b. (Reactor vessel/RCS [<i>PWR</i>] or RCP [<i>BWR</i>]) level less than (site-specific level). | CS1.1 | With CONTAINMENT CLOSURE not established, RCS level 64.5% RVLIS Full Range | 64.5% RVLIS Full Range corresponds to the level of six inches below the bottom ID of the RCS hot leg penetration (240' 7" el.). |
| 2 | a. CONTAINMENT CLOSURE established. AND b. (Reactor vessel/RCS [<i>PWR</i>] or RCP [<i>BWR</i>]) level less than (site-specific level). | CS1.2 | With CONTAINMENT CLOSURE established, RCS level < 59.8% RVLIS Full Range | 59.8% RVLIS Full Range corresponds to the top of active fuel. Other RCS level instruments are off-scale low when core uncover occurs. |
| 3 | a. (Reactor vessel/RCS [<i>PWR</i>] or RCP [<i>BWR</i>]) level cannot be monitored for 30 minutes or longer. AND b. Core uncover is indicated by ANY of the following: | CS1.3 | RCS water level cannot be monitored for ≥ 30 min. (Note 1) AND Core uncover is indicated by any of the following: <ul style="list-style-type: none"> • UNPLANNED increase in any Table C-1 sump or tank due to a loss of RCS | Added bulleted criteria "Visual observation of UNISOLABLE RCS leakage" to include direct observation of RCS leakage. Table C-1 lists site-specific applicable sumps and tanks. The dose rate due to this core shine should result in indications on installed area radiation monitors (R-32A or R-32B). If these radiation monitors reach and exceed 5 R/hr, a loss of inventory with potential to uncover the core is likely to have occurred. |

| | | | | |
|------|---|-----|---|--|
| | <ul style="list-style-type: none"> • (Site-specific radiation monitor) reading greater than (site-specific value) • Erratic source range monitor indication [<i>PWR</i>] • UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncover • (Other site-specific indications) | | <p>inventory</p> <ul style="list-style-type: none"> • Visual observation of unisolable RCS leakage • Containment High Range Radiation Monitor R-32A or R-32B > 5 Rem/hr • Erratic source range monitor indication | |
| Note | The Emergency Director should declare the Site Area Emergency promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded | N/A | Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|---|--------------------------|
| CG1 | Loss of (reactor vessel/RCS [<i>PWR</i>] or RCP [<i>BWR</i>]) inventory affecting fuel clad integrity with containment challenged MODE: Cold Shutdown, Refueling | CG1 | Loss of RCS inventory affecting fuel clad integrity with containment challenged MODE: Cold Shutdown, 6 - Refueling | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|---|
| 1 | a. (Reactor vessel/RCS [<i>PWR</i>] or RCP [<i>BWR</i>]) level less than (site-specific level) for 30 minutes or longer. AND b. ANY indication from the Containment Challenge Table (see below). | CG1.1 | RCS level < 59.8% RVLIS Full Range for ≥ 30 min. (Note 1) AND Any Containment Challenge indication, Table C-2 | 59.8% RVLIS Full Range corresponds to the top of active fuel. Other RCS level instruments are off-scale low when core uncover occurs. Table C-2 provides a tabularized list of containment challenge indications. 4% hydrogen concentration in the presence of oxygen represents an explosive mixture in containment. |
| 2 | a. (Reactor vessel/RCS [<i>PWR</i>] or RCP [<i>BWR</i>]) level cannot be monitored for 30 minutes or longer. AND b. Core uncover is indicated by ANY of the following: <ul style="list-style-type: none"> • (Site-specific radiation monitor) reading greater than (site-specific value) • Erratic source range monitor indication [<i>PWR</i>] | CG1.2 | RCS water level cannot be monitored for ≥ 30 min. (Note 1) AND Core uncover is indicated by any of the following: <ul style="list-style-type: none"> • UNPLANNED increase in any Table C-1 sump or tank due to a loss of RCS inventory • Visual observation of UNISOLABLE RCS leakage • Containment High Range | Added bulleted criteria “Visual observation of UNISOLABLE RCS leakage” to include direct observation of RCS leakage. The dose rate due to this core shine should result in indications on installed area radiation monitors (R-32A or R-32B). If these radiation monitors reach and exceed 5 R/hr, a loss of inventory with potential to uncover the core is likely to have occurred. Table C-1 lists site-specific applicable sumps and tanks. Table C-2 provides a tabularized list of containment challenge indications. 4% hydrogen concentration in the presence of oxygen represents an explosive mixture in containment. |

| | | | | |
|------|---|-----|---|---|
| | <ul style="list-style-type: none"> ● UNPLANNED increase in (site-specific sump and/or tank) levels of sufficient magnitude to indicate core uncover ● (Other site-specific indications) <p>AND</p> <p>c. ANY indication from the Containment Challenge Table (see below).</p> | | <p>Radiation Monitor R-32A or R-32B > 5 Rem/hr</p> <ul style="list-style-type: none"> ● Erratic source range monitor indication <p>AND</p> <p>Any Containment Challenge indication, Table C-2</p> | |
| Note | <p>The Emergency Director should declare the General Emergency promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.</p> <p>N/A</p> | N/A | <p>Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.</p> <p>Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is not required.</p> | <p>The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording.</p> <p>Note 6 implements the asterisked note associated with the generic Containment Challenge table.</p> |

| Containment Challenge Table |
|--|
| <ul style="list-style-type: none"> ■ 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 |
|--|
| <ul style="list-style-type: none">• CONTAINMENT CLOSURE not established (Note 6)• Containment hydrogen concentration \geq 4%• UNPLANNED rise in Containment pressure |

Category D

Permanently Defueled Station Malfunction

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|--|--|------------|----------------|---|
| PD-AU1 PD-AU2 PD-SU1 PD-HU1 PD-HU2 PD-HU3 PD-AA1 PD-AA2 PD-HA1 PD-HA3 | Recognition Category D Permanently Defueled Station | N/A | N/A | NEI Recognition Category PD ICs and EALs are applicable only to permanently defueled stations. RNP is not a defueled station. |

Category E

Independent Spent Fuel Storage Installation

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|---|--------------------------|
| E-HU1 | Damage to a loaded cask CONFINEMENT BOUNDARY MODE: All | EU1 | Damage to a loaded cask CONFINEMENT BOUNDARY | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|---|---|
| 1 | Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than (2 times the site-specific cask specific technical specification allowable radiation level) on the surface of the spent fuel cask. | EU1.1 | Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel > any Table E-1 dose limit | The values shown represent 2 times the limits specified in the 7P and 24P ISFSI Technical Specification for radiation external to a loaded HSM for a DSC. |

| Table E-1 ISFSI Dose Limits | |
|--|---|
| 7P ISFSI | 24P ISFSI |
| <ul style="list-style-type: none"> 400 mrem/hr outside of HSM door on centerline of DSC 400 mrem/hr at center of air inlets or outlets 100 mrem/hr on roof, front, back or side | <ul style="list-style-type: none"> 2,600 mrem/hr on the HSM front surface 10 mrem/hr on the HSM-H door centerline 20 mrem/hr on the end shield wall exterior |

Category F

Fission Product Barrier Degradation

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| FA1 | Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier. MODE: Power Operation, Hot Standby, Startup, Hot Shutdown | FA1 | Any loss or any potential loss of either Fuel Clad or RCS MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|--|
| 1 | Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier. | FA1.1 | Any loss or any potential loss of either Fuel Clad or RCS barrier (Table F-1) | Table F-1 provides the fission product barrier loss and potential loss thresholds. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| FS1 | Loss or Potential Loss of any two barriers MODE: Power Operation, Hot Standby, Startup, Hot Shutdown | FS1 | Loss or potential loss of any two barriers MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|--|
| 1 | Loss or Potential Loss of any two barriers | FS1.1 | Loss or potential loss of any two barriers (Table F-1) | Table F-1 provides the fission product barrier loss and potential loss thresholds. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| FG1 | Loss of any two barriers and Loss or Potential Loss of third barrier MODE: Power Operation, Hot Standby, Startup, Hot Shutdown | FG1 | Loss of any two barriers and loss or potential loss of the third barrier MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|--|--|
| 1 | Loss of any two barriers and Loss or Potential Loss of third barrier | FG1.1 | Loss of any two barriers AND Loss or potential loss of the third barrier (Table F-1) | Table F-1 provides the fission product barrier loss and potential loss thresholds. |

PWR Fuel Clad Fission Product Barrier Degradation Thresholds

| NEI FPB# | NEI Threshold Wording | RNP FPB #(s) | RNP FPB Wording | Difference Justification |
|--------------|---|-----------------|--|---|
| FC Loss 1 | RCS or SG Tube Leakage Not Applicable | N/A | N/A | N/A |
| FC Loss 2 | Inadequate Heat Removal A. Core exit thermocouple readings greater than (site-specific temperature value). | FC Loss B.1 | CSFST Core Cooling-RED Path entry conditions met | Consistent with the generic developers note options CSFST Core Cooling Red Path is used in lieu of CET temperatures. |
| FC Loss 3 | RCS Activity/CMNT Rad A. Containment radiation monitor reading greater than (site-specific value) OR B. (Site-specific indications that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131) | FC Loss C.1 | Containment High Range Radiation Monitor R-32A or R-32B > 100 R/hr | Containment radiation monitor readings greater than 100 Rem/hr indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the Containment. |
| | | FC Loss C.2 | Dose equivalent I-131 coolant activity > 300 µCi/gm | Coolant activity corresponding to > 300 µCi/gm is determined by sample analysis. Although RNP has a letdown radiation monitor (R-9), a threshold associated with this monitor is not implemented because in all credible instances in which an event resulting in 5% clad failure exists, letdown would be isolated. |
| FC Loss 4 | CNMT Integrity or Bypass Not Applicable | N/A | N/A | N/A |
| FC Loss 5 | Other Indications A. (site-specific as applicable) | N/A | N/A | No other site-specific Fuel Clad Loss indication has been identified for RNP. |

| NEI FPB# | NEI Threshold Wording | RNP FPB #(s) | RNP FPB Wording | Difference Justification |
|----------------|---|------------------|---|--|
| FC Loss 6 | ED Judgment A. ANY condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier. | FC Loss E.1 | Any condition in the opinion of the Emergency Coordinator that indicates loss of the fuel clad barrier | None |
| FC P-Loss 1 | RCS or SG Tube Leakage A. RCS/reactor vessel level less than (site-specific level) | N/A | N/A | See FC P-Loss B.1. The RCS level threshold is implemented as CSFST Core Cooling Orange Path conditions met. |
| FC P-Loss 2 | Inadequate Heat Removal A. Core exit thermocouple readings greater than (site-specific temperature value) OR B. Inadequate RCS heat removal capability via steam generators as indicated by (site-specific indications). | FC P-Loss B.1 | CSFST Core Cooling- ORANGE Path entry conditions met | Consistent with the generic developers note options CSFST Core Cooling Orange Path is used in lieu of CET temperatures. |
| | | FC P-Loss B.2 | CSFST Heat Sink- RED Path entry conditions met AND Heat sink is required | Consistent with the generic developers note options CSFST Heat Sink Red Path is used. The phrase “and heat sink required” was added to preclude the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. |
| FC P-Loss 3 | RCS Activity/CMNT Rad Not Applicable | N/A | N/A | N/A |
| FC P-Loss 4 | CNMT Integrity or Bypass Not Applicable | N/A | N/A | N/A |
| FC P-Loss 5 | Other Indications A. (site-specific as applicable) | N/A | N/A | No other site-specific Fuel Clad Potential Loss indication has been identified for RNP. |

| NEI FPB# | NEI Threshold Wording | RNP FPB #(s) | RNP FPB Wording | Difference Justification |
|-----------------------|--|---------------------|--|--------------------------|
| FC P-Loss 6 | Emergency Director Judgment A. Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier. | FC P-Loss E.1 | Any condition in the opinion of the Emergency Coordinator that indicates potential loss of the fuel clad barrier | None |

PWR RCS Fission Product Barrier Degradation Thresholds

| NEI FPB# | NEI IC Wording | RNP FPB #(s) | RNP FPB Wording | Difference Justification |
|---------------|--|-----------------|---|--|
| RCS Loss 1 | RCS or SG Tube Leakage A. An automatic or manual ECCS (SI) actuation is required by EITHER of the following: 1. UNISOLABLE RCS leakage OR 2. SG tube RUPTURE. | RCS Loss A.1 | An automatic or manual ECCS (SI) actuation required by EITHER : • UNISOLABLE RCS leakage • SG tube RUPTURE | None |
| RCS Loss 2 | Inadequate Heat Removal Not Applicable | N/A | N/A | N/A |
| RCS Loss 3 | RCS Activity/CMNT Rad A. Containment radiation monitor reading greater than (site-specific value). | RCS Loss C.1 | Containment High Range Radiation Monitor R-32A or R-32B > 5 R/hr | Containment radiation monitor readings greater than 5 R/hr indicate the release of reactor coolant to the Containment. |
| RCS Loss 4 | CNMT Integrity or Bypass Not Applicable | N/A | N/A | N/A |
| RCS Loss 5 | Other Indications A. (site-specific as applicable) | N/A | N/A | No other site-specific RCS Loss indication has been identified for RNP. |
| RCS Loss | Emergency Director Judgment A. ANY condition in the opinion | RCS Loss | Any condition in the opinion of the Emergency Coordinator | None |

| NEI FPB# | NEI IC Wording | RNP FPB #(s) | RNP FPB Wording | Difference Justification |
|-----------------|---|-------------------|--|--|
| 6 | of the Emergency Director that indicates Loss of the RCS Barrier. | E.1 | that indicates loss of the RCS barrier | |
| RCS P-Loss 1 | RCS or SG Tube Leakage A. Operation of a standby charging (makeup) pump is required by EITHER of the following: 1. UNISOLABLE RCS leakage OR 2. SG tube leakage. OR B. RCS cooldown rate greater than (site-specific pressurized thermal shock criteria/limits defined by site-specific indications). | RCS P-Loss A.1 | RCS leakage > capacity of a single charging pump (> 77 gpm) due to EITHER : <ul style="list-style-type: none"> • UNISOLABLE RCS leakage • SG tube leakage | Revised wording to capacity of a single charging pump (> 77 gpm). RNP runs 2 charging pumps in certain configurations. A leak > the capacity of a charging pump could exist without the starting of another charging pump being required. The proposed wording implements the intent of the generic threshold. |
| | | RCS P-Loss A.2 | CSFST Integrity- RED Path entry conditions met | Consistent with the generic developers note options CSFST Integrity Red Path is used. |
| RCS P-Loss 2 | Inadequate Heat Removal A. Inadequate RCS heat removal capability via steam generators as indicated by (site-specific indications). | RCS P-Loss B.1 | CSFST Heat Sink- RED Path conditions met AND Heat sink is required | Consistent with the generic developers note options CSFST Heat Sink Red Path is used. The phrase “and heat sink required” was added to preclude the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an EOP. |
| RCS P-Loss 3 | CS Activity/CMNT Rad Not Applicable | N/A | N/A | N/A |

| NEI FPB# | NEI IC Wording | RNP FPB #(s) | RNP FPB Wording | Difference Justification |
|-----------------|--|-------------------|---|---|
| RCS P-Loss 4 | CNMT Integrity or Bypass Not Applicable | N/A | N/A | N/A |
| RCS P-Loss 5 | Other Indications A. (site-specific as applicable) | N/A | N/A | No other site-specific RCS Potential Loss indication has been identified for RNP. |
| RCS P-Loss 6 | Emergency Director Judgment A. ANY condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier. | RCS P-Loss E.1 | Any condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier | None |

PWR Containment Fission Product Barrier Degradation Thresholds

| NEI FPB# | NEI IC Wording | RNP FPB #(s) | RNP FPB Wording | Difference Justification |
|-------------|--|---------------|---|--------------------------|
| CNMT Loss 1 | RCS or SG Tube Leakage A. A leaking or RUPTURED SG is FAULTED outside of containment. | CNMT Loss A.1 | A leaking or RUPTURED SG is FAULTED outside of containment | None |
| CNMT Loss 2 | Inadequate Heat Removal Not Applicable | N/A | N/A | N/A |
| CNMT Loss 3 | RCS Activity/CMNT Rad Not applicable | N/A | N/A | N/A |
| CNMT Loss 4 | CNMT Integrity or Bypass A. Containment isolation is required AND EITHER of the following: 1. Containment integrity has been lost based on Emergency Director judgment. OR 2. UNISOLABLE pathway from the containment to the environment exists. OR B. Indications of RCS leakage outside of containment. | CNMT Loss D.1 | Containment isolation is required AND EITHER: <ul style="list-style-type: none"> Containment integrity has been lost based on Emergency Coordinator judgment UNISOLABLE pathway from containment to the environment exists | None |
| | | CNMT Loss D.2 | Indications of RCS leakage outside of containment | None |

| NEI FPB# | NEI IC Wording | RNP FPB #(s) | RNP FPB Wording | Difference Justification |
|---------------------|---|-----------------------|---|---|
| CNMT Loss 5 | Other Indications A. (site-specific as applicable) | N/A | N/A | No other site-specific Containment Loss indication has been identified for RNP. |
| CNMT Loss 6 | Emergency Director Judgment ANY condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier. | CNMT Loss E.1 | Any condition in the opinion of the Emergency Coordinator that indicates loss of the containment barrier | None |
| CNMT P-Loss 1 | RCS or SG Tube Leakage Not Applicable | N/A | N/A | N/A |
| CNMT P-Loss 2 | Inadequate Heat Removal A. 1. (Site-specific criteria for entry into core cooling restoration procedure) AND 2. Restoration procedure not effective within 15 minutes. | CNMT P-Loss B.1 | CSFST Core Cooling- RED Path entry conditions met AND Restoration procedures not effective within 15 min. (Note 1) | Consistent with the generic developers note options CSFST Core Cooling Red Path is used in lieu of CET temperatures and RCS levels. Added Note 1 consistent with other thresholds with a timing component. |
| CNMT P-Loss 3 | RCS Activity/CMNT Rad A. Containment radiation monitor reading greater than (site-specific value). | CNMT P-Loss C.1 | Containment High Range Radiation Monitor R-32A or R-32B > 2000 R/hr | Containment radiation monitor readings greater than 2000 R/hr, indicate significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier. |
| CNMT P-Loss 4 | CNMT Integrity or Bypass A. Containment pressure greater than (site-specific value) | CNMT P-Loss D.1 | CSFST Containment- RED Path entry conditions me | Consistent with the generic developers note options CSFST Containment Red Path is used in lieu of containment pressure. |

| NEI FPB# | NEI IC Wording | RNP FPB #(s) | RNP FPB Wording | Difference Justification |
|-------------------------|---|---------------------------|--|--|
| | OR B. Explosive mixture exists inside containment OR C. 1. Containment pressure greater than (site-specific pressure setpoint) AND 2. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer. | CNMT P-Loss D.2 | Containment hydrogen concentration $\geq 4\%$ | 4% hydrogen concentration in the presence of oxygen represents an explosive mixture in containment. |
| | | CNMT P-Loss D.3 | Containment pressure ≥ 10 psig with < one full train of depressurization equipment operating (one Containment Spray System train AND one Containment Cooling System train) per design for > 15 min. (Note 1) | The Containment pressure setpoint (10 psig) is the pressure at which the containment depressurization systems should actuate. Limiting LOCA analyses assume one Containment Spray System train and one Containment Cooling System train operate. Added Note 1 consistent with other thresholds with a timing component. |
| CNMT P-Loss 5 | Other Indications A. (site-specific as applicable) | N/A | N/A | No other site-specific Containment Potential Loss indication has been identified for RNP. |
| CNMT P-Loss 6 | Emergency Director Judgment A. ANY condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier. | CNMT P-Loss E.1 | Any condition in the opinion of the Emergency Coordinator that indicates potential loss of the containment barrier | None |

Category H

Hazards and Other Conditions Affecting Plant Safety

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| HU1 | Confirmed SECURITY CONDITION or threat MODE: All | HU1 | Confirmed SECURITY CONDITION or threat. MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|--|--|
| 1 | A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the (site-specific security shift supervision). | HU1.1 | A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the Security 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. The Security Shift Supervision is defined as the Security Shift Supervision. |
| 2 | Notification of a credible security threat directed at the site. | | | |
| 3 | A validated notification from the NRC providing information of an aircraft threat. | | | |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|---|--------------------------|
| HU2 | Seismic event greater than OBE level MODE: All | HU2 | Seismic event greater than OBE level MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|--|--|
| 1 | Seismic event greater than Operating Basis Earthquake (OBE) as indicated by: (site-specific indication that a seismic event met or exceeded OBE limits) | HU2.1 | Seismic Recording Unit A or B indicates seismic event > Operating Basis Earthquake (0.1g horizontal OR 0.067g vertical) | Alarm lights on Seismic Recording Unit A and B provide indication of OBE exceedance. |

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

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|---|
| 1 | A tornado strike within the PROTECTED AREA. | HU3.1 | A tornado strike within the PROTECTED AREA | None |
| 2 | Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode. | HU3.2 | Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode | None |
| 3 | Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release). | HU3.3 | Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release) | None |
| 4 | A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles. | HU3.4 | A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7) | Added reference to Note 7. |
| 5 | (Site-specific list of natural or technological hazard events) | N/A | N/A | No other site-specific hazard has been identified for RNP. |
| Note | EAL #3 does not apply to routine traffic impediments such as fog, | N/A | Note 7: This EAL does not apply to routine traffic | This note, designated Note #7, is intended to apply to generic example EAL #4, not #3 as specified in the generic guidance. |

| | | | | |
|--|--|--|---|--|
| | snow, ice, or vehicle breakdowns or accidents. | | impediments such as fog, snow, ice, or vehicle breakdowns or accidents. | |
|--|--|--|---|--|

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| HU4 | FIRE potentially degrading the level of safety of the plant. MODE: All | HU4 | FIRE potentially degrading the level of safety of the plant MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|---|--|
| 1 | <p>a. A FIRE is NOT extinguished within 15-minutes of ANY of the following FIRE detection indications:</p> <ul style="list-style-type: none"> ● 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 <p>AND</p> <p>b. The FIRE is located within ANY of the following plant rooms or areas: (site-specific list of plant rooms or areas)</p> | HU4.1 | <p>A FIRE is not extinguished within 15 min. of any of the following FIRE detection indications (Note 1):</p> <ul style="list-style-type: none"> ● 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 <p>AND</p> <p>The FIRE is located within any Table H-1 area</p> | Table H-1 provides a tabularized list of site-specific fire areas. |
| 2 | <p>a. Receipt of a single fire alarm (i.e., no other indications of a FIRE).</p> <p>AND</p> <p>b. The FIRE is located within</p> | HU4.2 | <p>Receipt of a single fire alarm (i.e., no other indications of a FIRE)</p> <p>AND</p> <p>The fire alarm is indicating a</p> | Table H-1 provides a list of site-specific fire areas. |

| | | | | |
|------|---|-------|---|---|
| | <p>ANY of the following plant rooms or areas: (site-specific list of plant rooms or areas)</p> <p>AND</p> <p>c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.</p> | | <p>FIRE within any Table H-1 area</p> <p>AND</p> <p>The existence of a FIRE is not verified within 30 min. of alarm receipt (Note 1)</p> | |
| 3 | <p>A FIRE within the plant <i>or ISFSI</i> [for plants with an ISFSI outside the plant Protected Area] PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.</p> | HU4.3 | <p>A FIRE within the plant PROTECTED AREA not extinguished within 60 min. of the initial report, alarm or indication (Note 1)</p> | RNP does not have an ISFSI outside the Protected Area. |
| 4 | <p>A FIRE within the plant <i>or ISFSI</i> [for plants with an ISFSI outside the plant Protected Area] PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.</p> | HU4.4 | <p>A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish</p> | RNP does not have an ISFSI outside the Protected Area. |
| Note | <p>Note: The Emergency Director should declare the Unusual Event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.</p> | N/A | <p>Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.</p> | <p>The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording.</p> |

| Table H-1 Fire Areas |
|--|
| <ul style="list-style-type: none">• Containment• Auxiliary Building• Control Room• Fuel Handling Building• Intake• AFW Room• 4 KV Switchgear Room• E-1/E-2 Switchgear Room• RWST• CST |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| HU7 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of a (NO)UE MODE: All | HU7 | Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a UE MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|--|--------------------------|
| 1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs. | HU7.1 | Other conditions exist which in the judgment of the Emergency Coordinator 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 | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| HA1 | HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes. MODE: All | HA1 | HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|--|--|
| 1 | A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the (site-specific security shift supervision). | HA1.1 | A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift 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. The Security Shift Supervision is the site-specific security shift supervision. |
| 2 | A validated notification from NRC of an aircraft attack threat within 30 minutes of the site. | | | |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|--|--------------------------|
| HA5 | Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown. MODE: All | N/A | Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown. MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|--|--|
| 1 | a. Release of a toxic, corrosive, asphyxiant or flammable gas into any of the following plant rooms or areas: (site-specific list of plant rooms or areas with entry-related mode applicability identified) AND b. Entry into the room or area is prohibited or impeded. | HA5.1 | Release of a toxic, corrosive, asphyxiant or flammable gas into any Table R-2/H-2 rooms or areas AND Entry into the room or area is prohibited or IMPEDED (Note 5) | Table R-2/H-2 provides a list of safe shutdown rooms/areas and applicable operating modes. |
| Note | Note: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted. | Note 5 | If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted. | None |

| Table R-2/H-2 Safe Operation & Shutdown Rooms/Areas | |
|---|----------------|
| Room/Area | Mode(s) |
| Reactor Auxiliary Building, 1 st level hallway | 1,2,3,4,5 |
| Reactor Auxiliary Building, 2 nd level hallway | 1,2,3,4,5 |
| Charging Pump Room | 1,2,3,4,5 |
| Component Cooling Water Pump Room | 1,2,3,4,5 |
| Primary Sample Room | 1,2,3,4,5 |
| Primary Demineralizer Room | 1,2,3 |
| Spent Fuel Pump / Heat Exchanger Room | 1,2,3,4,5 |
| Pipe Alley | 4 |
| RHR Heat Exchanger Room | 4 |
| RHR Pump Room entry area (access to RHR Pump CCW flow indication / control) | 4 |
| Boric Acid Batch Tank Room | 1,2,3,4,5 |
| Emergency Bus E1/E2 Room | 3,4,5 |
| Turbine Building 1 st Floor (includes Condensate Polisher, Makeup Water Treatment and Secondary Sample Room) | 1,2,3,4 |
| Turbine Building 2 nd Floor | 1,2,3,4 |
| Turbine Building 3 rd Floor | 1,3,4 |
| Containment Building | 3 |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| HA6 | Control Room evacuation resulting in transfer of plant control to alternate locations. MODE: All | HA6 | Control Room evacuation resulting in transfer of plant control to alternate locations MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|---|
| 1 | An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations). | HA6.1 | An event has resulted in plant control being transferred from the Control Room to the Dedicated/Alternate Shutdown System | Dedicated/Alternate Shutdown System is the site-specific remote shutdown panels/local control stations. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|---|--------------------------|
| HA7 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert. MODE: All | HA7 | Other conditions exist that in the judgment of the Emergency Coordinator warrant declaration of an Alert MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|--------------------------|
| 1 | Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels. | HA7.1 | Other conditions exist which, in the judgment of the Emergency Coordinator, 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 | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|---|--------------------------|
| HS1 | HOSTILE ACTION within the PROTECTED AREA MODE: All | HS1 | HOSTILE ACTION within the PROTECTED AREA MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|---|
| 1 | A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision). | HS1.1 | A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision | The Security Shift Supervision is the site-specific security shift supervision. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|---|--------------------------|
| HS6 | Inability to control a key safety function from outside the Control Room. MODE: All | HS6 | Inability to control a key safety function from outside the Control Room MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|---|
| 1 | <p>a. An event has resulted in plant control being transferred from the Control Room to (site-specific remote shutdown panels and local control stations).</p> <p>AND</p> <p>b. Control of ANY of the following key safety functions is not reestablished within (site-specific number of minutes).</p> <ul style="list-style-type: none"> ● Reactivity control ● Core cooling [<i>PWR</i>] / RCP water level [<i>BWR</i>] ● RCS heat removal | HS6.1 | <p>An event has resulted in plant control being transferred from the Control Room to the Dedicated/Alternate Shutdown System</p> <p>AND</p> <p>Control of any of the following key safety functions is not reestablished within 15 min. (Note 1):</p> <ul style="list-style-type: none"> ● Reactivity ● Core cooling ● RCS heat removal | The Dedicated/Alternate Shutdown System is the site-specific remote shutdown panels/local control stations. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|---|--------------------------|
| HS7 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency. MODE: All | HS7 | Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a Site Area Emergency MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|--|--------------------------|
| 1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary. | HS7.1 | Other conditions exist which in the judgment of the Emergency Coordinator 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 | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|---|--------------------------|
| HG1 | HOSTILE ACTION resulting in loss of physical control of the facility. MODE: All | HG1 | HOSTILE ACTION resulting in loss of physical control of the facility MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|---|---|
| 1 | <p>a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-specific security shift supervision).</p> <p>AND</p> <p>b. EITHER of the following has occurred:</p> <ol style="list-style-type: none"> ANY of the following safety functions cannot be controlled or maintained. <ul style="list-style-type: none"> Reactivity control Core cooling [PWR]/RCP water level [BWR] RCS heat removal <p>OR</p> <ol style="list-style-type: none"> Damage to spent fuel has occurred or is IMMINENT. | HG1.1 | <p>A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision</p> <p>AND EITHER of the following has occurred:</p> <p>Any of the following safety functions cannot be controlled or maintained</p> <ul style="list-style-type: none"> Reactivity control Core cooling RCS heat removal <p>OR</p> <p>Damage to spent fuel has occurred or is IMMINENT</p> | The Security Shift Supervision is the site-specific security shift supervision. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|---|--------------------------|
| HG7 | Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency MODE: All | HG7 | Other conditions exist which in the judgment of the Emergency Coordinator warrant declaration of a General Emergency MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|--|--------------------------|
| 1 | Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area. | HG7.1 | Other conditions exist which in the judgment of the Emergency Coordinator 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 | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|---|--------------------------|
| SU1 | Loss of all offsite AC power capability to emergency buses for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU1 | Loss of all offsite AC power capability to emergency buses for 15 minutes or longer MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|--|
| 1 | Loss of ALL offsite AC power capability to (site-specific emergency buses) for 15 minutes or longer. | SU1.1 | Loss of all offsite AC power capability to 480V emergency buses E-1 and E-2 for ≥ 15 min. (Note 1) | 480V buses E-1 and E-2 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 Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|---|--------------------------|
| SU2 | UNPLANNED loss of Control Room indications for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU3 | UNPLANNED loss of Control Room indications for 15 minutes or longer. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|---|--|
| 1 | An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer. | SU3.1 | An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1) | The site-specific Safety System Parameters 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 Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

| [BWR parameter list] | [PWR parameter list] |
|------------------------------|--|
| Reactor Power | Reactor Power |
| RCP Water Level | RCS Level |
| RCP Pressure | RCS Pressure |
| Primary Containment Pressure | In-Core/Core Exit Temperature |
| Suppression Pool Level | Levels in at least (site-specific number) steam generators |
| Suppression Pool Temperature | Steam Generator Auxiliary or Emergency Feed Water Flow |

| Table S-1 Safety System Parameters |
|--|
| <ul style="list-style-type: none">• Reactor power• RCS level• RCS pressure• Core exit T/C temperature• Level in at least one SG• Auxiliary feed flow in at least one SG |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|---|---|
| SU3 | Reactor coolant activity greater than Technical Specification allowable limits. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU4 | RCS activity greater than Technical Specification allowable limits MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby | Changed 'reactor coolant activity" to "RCS activity" to conform to site specific terminology. Consistent with the Technical Specification 3.4.16 RCS activity limit applicability, this EAL is only applicable in Modes 1, 2 and Mode 3 when RCS temperature is $\geq 500^{\circ}\text{F}$ |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|---|--|
| 1 | (Site-specific radiation monitor) reading greater than (site-specific value). | SU4.2 | With letdown in service, letdown line area radiation monitor R-9 > 500 mR/hr (Note 9) | Downstream of the non-regenerative heat exchanger and upstream of the mixed bed demineralizers, the letdown stream passes by area radiation monitor R-9, which is mounted above the letdown line pipe. In order for R-9 readings to represent fission product activity in the reactor coolant and thereby warn of potential fuel element failure, letdown must be in service allowing flow through the letdown line and past the radiation monitor. The threshold value of 500 mR/hr represents fuel failure in excess of 0.1%. |
| 2 | Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications. | SU4.1 | RCS activity > Technical Specification Section 3.4.16 limits (Note 9) | Changed 'reactor coolant activity" to "RCS activity" to conform to site specific terminology. RNP T.S. Section 3.4.16 provides the TS allowable coolant activity limits. |
| N/A | N/A | N/A | Note 9: Mode 3 applicable only when RCS temperature is $\geq 500^{\circ}\text{F}$ | Consistent with the Technical Specification 3.4.16 RCS activity limit applicability, this EAL is only applicable in Modes 1, 2 and Mode 3 when RCS temperature is $\geq 500^{\circ}\text{F}$ |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|---|--------------------------|
| SU4 | RCS leakage for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU5 | RCS leakage for 15 minutes or longer MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|--|--|
| 1 | RCS unidentified or pressure boundary leakage greater than (site-specific value) for 15 minutes or longer. | SU5.1 | RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. OR RCS identified leakage > 25 gpm for ≥ 15 min. OR Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min. (Note 1) | Example EALs #1, 2 and 3 have been combined into a single EAL. |
| 2 | RCS identified leakage greater than (site-specific value) for 15 minutes or longer. | | | |
| 3 | Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer. | | | |
| Note | The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded. | N/A | Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|---|--|
| SU5 | Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor. MODE: Power Operation | SU6 | Automatic or manual trip fails to shut down the reactor MODE: 1 - Power Operation, 2 - Startup | Added Mode 2 Startup applicability. Consistent with the generic developer notes, the shutdown criteria of reactor power = 5% being Mode 2 for RNP, requires inclusion of Mode 2 applicability. |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|--|--|
| 1 | a. An automatic (trip [PWR] / scram [BWR]) did not shutdown the reactor. AND b. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor. | SU6.1 | An automatic trip did not shut down the reactor as indicated by reactor power \geq 5% after any RPS setpoint is exceeded AND A subsequent automatic trip or manual trip action taken at the RTGB is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8) | As specified in the generic developers guidance "Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level)." Reactor power < 5% is the site-specific indication of a successful reactor trip. Added the words "... as indicated by reactor power \geq 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. Actions taken at the RTGB (reactor trip pushbuttons) are the site-specific actions credited for a successful manual trip. |
| 2 | a. A manual trip ([PWR] / scram [BWR]) did not shutdown the reactor. AND b. EITHER of the following: 1. A subsequent manual action taken at the reactor control consoles is successful in shutting down the reactor. | SU6.2 | A manual trip did not shut down the reactor as indicated by reactor power \geq 5% after any manual trip action was initiated AND A subsequent automatic trip or manual trip action taken at the RTGB is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8) | As specified in the generic developers guidance "Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level)." Reactor power < 5% is the site-specific indication of a successful reactor trip. Added the words "... as indicated by reactor power \geq 5% after any manual trip action was initiated" to clarify that it is a failure of any manual trip when an actual manual trip signal has been inserted. Combined conditions b.1 and b.2 into a single statement to simplify the presentation. Actions taken at the RTGB (reactor trip pushbuttons) are the site- |

| | | | | |
|-------|--|-----|---|---|
| | OR 2 A subsequent automatic (trip [PWR] / scram [BWR]) is successful in shutting down the reactor. | | | specific actions credited for a successful manual trip. |
| Notes | Note: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies. | N/A | Note 8: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies. | None |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| SU6 | Loss of all onsite or offsite communications capabilities. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU7 | Loss of all onsite or offsite communications capabilities. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|---|---|
| 1 | Loss of ALL of the following onsite communication methods: (site-specific list of communications methods) | SU7.1 | Loss of all Table S-3 onsite communication methods OR Loss of all Table S-3 offsite communication methods OR Loss of all Table S-3 NRC communication methods | Example EALs #1, 2 and 3 have been combined into a single EAL. Table S-3 provides a site-specific list of onsite, offsite and NRC communications methods. Replaced the acronym "ORO" with the word "offsite". ORO is not a standard acronym. |
| 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-3 Communication Methods | | | |
|--|---------------|------------|------------|
| System | Onsite | ORO | NRC |
| Public Address System | X | | |
| PBX Telephone System | X | | |
| Radio Transceivers for RNP and Vicinity | X | | |
| Back-up Telephone System (ESSX) | X | | |
| Plant Security Radio Transceivers | X | | |
| Corporate Telephone Communications System (Voicenet) | | X | X |
| BellSouth | | X | X |
| Dedicated Telephone System to Load Dispatcher | | X | |
| Plant Security Radio Control Station | | X | |
| DEMNET | | X | |
| NRC Emergency Telecommunication System (ETS) | | | X |
| Satellite Phones | | X | X |
| Cellular Phones | | X | X |
| Palmetto 800 Transceivers | | X | |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|---|--------------------------|
| SU7 | Failure to isolate containment or loss of containment pressure control. [PWR] MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SU8 | Failure to isolate containment or loss of containment pressure control MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|---|
| 1 | a. Failure of containment to isolate when required by an actuation signal. AND b. ALL required penetrations are not closed within 15 minutes of the actuation signal. | SU8.1 | EITHER: Any penetration is not isolated within 15 min. of a VALID containment isolation signal OR Containment pressure ≥ 10 psig with < one full train of depressurization equipment operating (one Containment Spray System train AND one Containment Cooling System train) per design for ≥ 15 min. (Note 1) | Reworded EAL to better describe the intent. Penetrations cannot close, but they can be isolated by closure of one or more isolation valves associated with that penetration. The revised wording maintains the generic example EAL intent while more clearly describing failure to isolate threshold. The Containment pressure setpoint (10 psig) is the pressure at which the containment depressurization systems should actuate. Limiting LOCA analyses assume one Containment Spray System train and one Containment Cooling System train operate. |
| 2 | a. Containment pressure greater than (site-specific pressure). AND b. Less than one full train of (site-specific system or equipment) is operating per design for 15 minutes or longer. | | | |
| N/A | N/A | N/A | Note 1: The Emergency Coordinator should declare the event | Added Note 1 to be consistent in its use for EAL thresholds with a timing component. |

| | | | | |
|--|--|--|---|--|
| | | | <div>promptly upon determining that time limit has been exceeded, or will likely be exceeded.</div> | |
|--|--|--|---|--|

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|---|--------------------------|
| SA1 | <p>Loss of all but one AC power source to emergency buses for 15 minutes or longer.</p> <p>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown</p> | SA1 | <p>Loss of all but one AC power source to emergency buses for 15 minutes or longer.</p> <p>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown</p> | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|--|--|
| 1 | <p>a. AC power capability to (site-specific emergency buses) is reduced to a single power source for 15 minutes or longer.</p> <p>AND</p> <p>b. Any additional single power source failure will result in a loss of all AC power to SAFETY SYSTEMS.</p> | SA1.1 | <p>AC power capability to 480V emergency buses E-1 and E-2 reduced to a single power source for ≥ 15 min. (Note 1)</p> <p>AND</p> <p>Any additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS</p> | 480V buses E-1 and E-2 are the site-specific emergency buses. |
| 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 Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|---|--------------------------|
| SA2 | <p>UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.</p> <p>MODE: Power Operation, Startup, Hot Standby, Hot Shutdown</p> | SA3 | <p>UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.</p> <p>MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown</p> | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|--|--|
| 1 | <p>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.</p> <p>AND</p> <p>ANY of the following transient events in progress.</p> <ul style="list-style-type: none"> Automatic or manual runback greater than 25% thermal reactor power Electrical load rejection greater than 25% full electrical load Reactor scram [<i>BWR</i>] / trip [<i>PWR</i>] ECCS (SI) actuation Thermal power oscillations greater than 10% [<i>BWR</i>] | SA3.1 | <p>An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1)</p> <p>AND</p> <p>Any significant transient is in progress, Table S-2</p> | <p>The site-specific Safety System Parameters are listed in Table S-1.</p> <p>The site-specific significant transients are listed in Table S-2.</p> <p>RNP is a PWR and thus does not include thermal power oscillations > 10%.</p> |

| | | | | |
|------|--|-----|---|--|
| 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 Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |
|------|--|-----|---|--|

| <i>[BWR parameter list]</i> | <i>[PWR parameter list]</i> |
|------------------------------|--|
| Reactor Power | Reactor Power |
| RCP Water Level | RCS Level |
| RCP Pressure | RCS Pressure |
| Primary Containment Pressure | In-Core/Core Exit Temperature |
| Suppression Pool Level | Levels in at least (site-specific number) steam generators |
| Suppression Pool Temperature | Steam Generator Auxiliary or Emergency Feed Water Flow |

Table S-1 Safety System Parameters

- Reactor power
- RCS level
- RCS pressure
- Core exit T/C temperature
- Level in at least one SG
- Auxiliary feed flow in at least one SG

Table S-2 Significant Transients

- Reactor trip
- Runback > 25% thermal power
- Electrical load rejection > 25% electrical load
- Safety injection actuation

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|---|--|
| SA5 | Automatic or manual (trip [PWR] / scram [BWR]) fails to shutdown the reactor, and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor. MODE: Power Operation | SA6 | Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor MODE: 1 - Power Operation, 2 - Startup | Added Mode 2 Startup applicability. Consistent with the generic developer notes, the shutdown criteria of reactor power = 5% being Mode 2 for RNP, requires inclusion of Mode 2 applicability. |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|---|
| 1 | a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor. AND b. Manual actions taken at the reactor control consoles are not successful in shutting down the reactor. | SA6.1 | An automatic or manual trip fails to shut down the reactor as indicated by reactor power $\geq 5\%$ AND Manual trip actions taken at the RTGB are not successful in shutting down the reactor as indicated by reactor power $\geq 5\%$ (Note 8) | As specified in the generic developers guidance "Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level)." Reactor power < 5% is the site-specific indication of a successful reactor trip. Actions taken at the RTGB (reactor trip pushbuttons) are the site-specific actions credited for a successful manual trip. |
| Notes | Note: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies. | N/A | Note 8: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron | None |

| | | | | |
|--|--|--|-----------------------|--|
| | | | injection strategies. | |
|--|--|--|-----------------------|--|

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| SA9 | Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SA9.1 | Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode MODE: All | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|------------------|--|--------------|---|---|
| 1 | <p>a. The occurrence of ANY of the following hazardous events:</p> <ul style="list-style-type: none"> ● 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 <p>AND</p> <p>b. EITHER of the following:</p> <ol style="list-style-type: none"> 1. Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode. <p>OR</p> <ol style="list-style-type: none"> 2. The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode. | SA9.1 | <p>The occurrence of any Table S-4 hazardous event</p> <p>AND EITHER:</p> <ul style="list-style-type: none"> ● Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode ● The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode | The hazardous events have been listed in Table S-4. |

| Table S-4 Hazardous Events |
|--|
| <ul style="list-style-type: none">● 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 | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| SS1 | Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SS1 | Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|--|
| 1 | Loss of ALL offsite and ALL onsite AC power to (site-specific emergency buses) for 15 minutes or longer. | SS1.1 | Loss of all offsite and all onsite AC power capability to 480V emergency buses E-1 and E-2 for ≥ 15 min. (Note 1) | 480V buses E-1 and E-2 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 Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--|
| SS5 | Inability to shutdown the reactor causing a challenge to (core cooling [PWR] / RCP water level [BWR]) or RCS heat removal. MODE: Power Operation | SS6 | Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal MODE: 1 - Power Operation, 2 - Startup | Added Mode 2 Startup applicability. Consistent with the generic developer notes, the shutdown criteria of reactor power = 5% being Mode 2 for RNP, requires inclusion of Mode 2 applicability. |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|--|---|
| 1 | <p>a. An automatic or manual (trip [PWR] / scram [BWR]) did not shutdown the reactor.</p> <p>AND</p> <p>b. All manual actions to shutdown the reactor have been unsuccessful.</p> <p>AND</p> <p>c. EITHER of the following conditions exist:</p> <ul style="list-style-type: none"> • (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 | <p>An automatic or manual trip fails to shut down the reactor as indicated by reactor power $\geq 5\%$</p> <p>AND</p> <p>All actions to shut down the reactor are not successful as indicated by reactor power $\geq 5\%$</p> <p>AND EITHER:</p> <ul style="list-style-type: none"> • Core Cooling RED Path entry conditions met • Heat Sink RED Path entry conditions met | <p>As specified in the generic developers guidance "Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level)." Reactor power < 5% is the site-specific indication of a successful reactor trip.</p> <p>Indication that core cooling is extremely challenged is manifested by CSFST Core Cooling RED Path entry conditions met.</p> <p>Indication that heat removal is extremely challenged is manifested by CSFST Heat Sink RED Path entry conditions met.</p> |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|---|------------|--|--------------------------|
| SS8 | Loss of all Vital DC power for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SS2 | Loss of all vital DC power for 15 minutes or longer. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown | None |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|---|
| 1 | Indicated voltage is less than (site-specific bus voltage value) on ALL (site-specific Vital DC busses) for 15 minutes or longer. | SS2.1 | Loss of all vital DC power based on < 109.5 VDC Bus A and < 106.2 VDC Bus B voltage indications for ≥ 15 min. (Note 1) | 109.5 VDC Bus A and 106.2 VDC Bus B are the calculated site-specific minimum vital DC bus voltages. DC buses A and B are the site-specific vital DC buses. |
| Note | The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded. | N/A | Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|---|--|
| SG1 | Prolonged loss of all offsite and all onsite AC power to emergency buses. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SG1a | Prolonged loss of all offsite and all onsite AC power to emergency buses MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown | NEI ICs SG1 and SG8 are grouped under the same RNP IC category for simplification. The RNP emergency buses are the site-specific emergency buses. |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|--|-----------|---|--|
| 1 | a. Loss of ALL offsite and ALL onsite AC power to (site-specific emergency buses). AND b. EITHER of the following: <ul style="list-style-type: none"> 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 480V emergency buses E-1 and E-2 AND EITHER: <ul style="list-style-type: none"> Restoration of at least one emergency bus in < 8 hours is not likely (Note 1) Core Cooling RED Path entry conditions met | 480V buses E-1 and E-2 are the site-specific emergency buses. 8 hours is the site-specific SBO coping analysis time. CSFST Core Cooling RED Path entry conditions met indicates significant core exit superheating and core uncovery. |
| Note | The Emergency Director should declare the General Emergency promptly upon determining that (site-specific hours) has been exceeded, or will likely be exceeded. | N/A | Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

| NEI IC# | NEI IC Wording | RNP IC#(s) | RNP IC Wording | Difference Justification |
|---------|--|------------|--|--|
| SG8 | Loss of all AC and Vital DC power sources for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown | SG1b | Loss of all AC and vital DC power sources for 15 minutes or longer MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown | NEI ICs SG1 and SG8 are grouped under the same RNP IC category for simplification. |

| NEI Ex. EAL # | NEI Example EAL Wording | RNP EAL # | RNP EAL Wording | Difference Justification |
|---------------|---|-----------|---|--|
| 1 | a. Loss of ALL offsite and ALL onsite AC power to (site-specific emergency buses) for 15 minutes or longer. AND b. Indicated voltage is less than (site-specific bus voltage value) on ALL (site-specific Vital DC busses) for 15 minutes or longer. | SG1.2 | Loss of all offsite and all onsite AC power capability to 480V emergency buses E-1 and E-2 for ≥ 15 min. AND Loss of all vital DC power based on < 109.5 VDC Bus A and < 106.2 VDC Bus B voltage indications for ≥ 15 min. (Note 1) | 480V buses E-1 and E-2 are the site-specific emergency buses. 109.5 VDC Bus A and 106.2 VDC Bus B are the calculated site-specific minimum vital DC bus voltages. DC buses A and B are the site-specific vital DC buses. |
| Note | The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded. | N/A | Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. | The classification timeliness note has been standardized across the RNP EAL scheme by referencing the "time limit" specified within the EAL wording. |

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Enclosure 3
278 Pages (including cover page)

Enclosure 3

EMERGENCY ACTION LEVEL TECHNICAL BASES DOCUMENT (CLEAN
VERSION)



Robinson Nuclear Plant

I
Information
Use

H. B. Robinson Steam Electric Plant
Plant Operating Manual
Volume 2
Part 5

EPCLA-04

***EMERGENCY ACTION LEVEL TECHNICAL
BASES DOCUMENT***

(Clean Version)

Revision 0 Draft 4/24/15

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1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Robinson Nuclear Plant (RNP). It should be used to facilitate review of the RNP EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EPCLA-01-110, Emergency Control, may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Coordinator in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training and for explaining event classifications to off-site officials.

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

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

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 RNP 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 (ADAMS Accession Number ML12326A805) (ref. 4.1.1), RNP conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

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

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

The primary fission product barriers are:

- A. Fuel Clad: The Fuel Clad Barrier is the zircalloy tubes that contain the fuel pellets.
- B. Reactor Coolant System (RCS): The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.
- C. Containment: The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency

2.3 Fission Product Barrier Classification Criteria

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

Alert:

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

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

2.4 EAL Organization

The RNP EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
 - EALs applicable under all plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
 - EALs applicable only under hot operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Hot Standby, Startup, or Power Operation mode.
 - EALs applicable only under cold operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling or Defueled mode.

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

- Within each 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 RNP EAL categories are aligned to and represent the NEI 99-01 "Recognition Categories." Subcategories are used in the RNP scheme as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The RNP EAL categories and subcategories are listed below.

EAL Groups, Categories and Subcategories

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

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

2.5 Technical Bases Information

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

Category Letter & Title

Subcategory Number & Title

Initiating Condition (IC)

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

EAL Identifier (enclosed in rectangle)

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

1. First character (letter): Corresponds to the EAL category as described above (R, C, H, S, F or E)
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 Operations, 2 - Startup, 3 – Hot Standby, 4 - Hot Shutdown, 5

- Cold Shutdown, 6 - Refueling, D - Defueled, or All. (See Section 2.6 for operating mode definitions)

Definitions:

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

Basis:

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

RNP Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.6 Operating Mode Applicability (ref. 4.1.7)

1 Power Operations

$K_{\text{eff}} \geq 0.99$ and reactor thermal power $> 5\%$ and average coolant temperature $\geq 350^\circ\text{F}$

2 Startup

$K_{\text{eff}} \geq 0.99$ and reactor thermal power $\leq 5\%$ average coolant temperature $\geq 350^\circ\text{F}$

3 Hot Standby

$K_{\text{eff}} < 0.99$ and average coolant temperature $\geq 350^\circ\text{F}$

3 Hot Shutdown

$K_{\text{eff}} < 0.99$ and average coolant temperature $350^\circ\text{F} > T_{\text{avg}} > 200^\circ\text{F}$ (excluding decay heat)

4 Cold Shutdown

$K_{\text{eff}} < 0.99$ and average coolant temperature $T_{\text{avg}} \leq 200^\circ\text{F}$

5 Refueling

$K_{\text{eff}} < 0.95$ and average coolant temperature $T_{\text{avg}} \leq 140^\circ\text{F}$; fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed

D Defueled

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

The plant operating mode that exists at the time that the event occurs (prior to any protective system or operator action 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 Coordinator must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes, and the informing basis information. In the Recognition Category F matrices, EALs are based on loss or potential loss of Fission Product Barrier Thresholds.

3.1.1 Classification Timeliness

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, "Interim Staff Guidance, Emergency Planning for Nuclear Power Plants" (ref. 4.1.9).

3.1.2 Valid Indications

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

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

3.1.3 Imminent Conditions

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

3.1.4 Planned vs. Unplanned Events

A planned work activity that results in an expected event or condition which meets or exceeds an EAL does not warrant an emergency declaration provided that: 1) the activity proceeds as planned, and 2) the plant remains within the limits imposed by the operating license. Such activities include planned work to test, manipulate, repair, maintain or modify a system or

component. In these cases, the controls associated with the planning, preparation and execution of the work will ensure that compliance is maintained with all aspects of the operating license provided that the activity proceeds and concludes as expected. Events or conditions of this type may be subject to the reporting requirements of 10 § CFR 50.72 (ref. 4.1.4).

3.1.5 Classification Based on Analysis

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

3.1.6 Emergency Coordinator 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 Coordinator 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 Coordinator 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. 4.1.14).

3.2.1 Classification of Multiple Events and Conditions

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

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

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

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

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

3.2.2 Consideration of Mode Changes During Classification

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

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

3.2.3 Classification of Imminent Conditions

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

3.2.4 Emergency Classification Level Upgrading and Downgrading

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

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

3.2.5 Classification of Short-Lived Events

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

3.2.6 Classification of Transient Conditions

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

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

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

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

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

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

In some cases, an EAL may be met but the emergency classification was not made at the time of the event or condition. This situation can occur when personnel discover that an event or condition existed which met an EAL, but no emergency was declared, and the event or

condition no longer exists at the time of discovery. This may be due to the event or condition not being recognized at the time or an error that was made in the emergency classification process.

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 (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 Drawing HBR2-9800, Plot Plan RNP
- 4.1.7 Technical Specifications Table 1.1-1 Modes
- 4.1.8 PRO-NGGC-0201 NGG Procedure Writers Guide
- 4.1.9 NSIR/DPR-ISG-01 Interim Staff Guidance, Emergency Planning for Nuclear Power Plants
- 4.1.10 PLP-007 Robinson Emergency Plan
- 4.1.11 UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone
- 4.1.12 OMP-003, Shutdown Safety Function Guidelines
- 4.1.13 OMM-033, Implementation of CV Closure
- 4.1.14 CM-603, Disassembly and Assembly of the Containment Equipment Hatch and Missile Barrier

4.2 Implementing

- 4.2.1 EPCLA-01, Emergency Control
- 4.2.2 NEI 99-01 Rev. 6 to RNP EAL Comparison Matrix
- 4.2.3 RNP 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 process, or have occurred, which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of hostile action. Any releases are expected to be small fractions of the EPA Protective Action Guideline exposure levels.

Confinement Boundary

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.

Containment Closure

The action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. In accordance with OMM-033, Implementation of CV Closure, Containment closure exits when (ref. 4.1.13, 4.1.14):

- The equipment hatch is closed and held in place by a sufficient number of bolts as identified in CM-603, AND
- At least one door in the personnel air lock is closed, AND
- Each penetration providing direct access from the Containment atmosphere to the outside atmosphere is either:
 - closed by a manual or automatic isolation valve, blind flange, or equivalent,
 - OR
 - capable of being closed by an OPERABLE Containment Ventilation Isolation System.

EPA PAGs

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

Explosion

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

Faulted

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

Fire

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

Flooding

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

General Emergency

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

Hostage

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

Hostile Action

An act toward RNP 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 RNP. 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.

Maintain

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

Normal Levels

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

Owner Controlled Area

That area surrounding the Protected Area beyond which RNP exercises access control.

Projectile

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

Protected Area

An area encompassed by physical barriers and to which access is controlled.
The Protected Area refers to the designated Security area around the process buildings and is depicted in Drawing HBR2-9800, Plot Plan RNP (ref. 4.1.6).

RCS Intact

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

Refueling Pathway

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

Reduced Inventory

Plant condition when fuel is in the Reactor Vessel and Reactor Coolant System level is less than or equal to -36 inches below the vessel flange (ref. 4.1.12, 4.1.13).

Ruptured

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

Restore

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

Safety System

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

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

- (1) The integrity of the reactor coolant pressure boundary;
- (2) The capability to shut down the reactor and maintain it in a safe shutdown condition;
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Security Condition

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

Site Boundary

As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone (ref. 4.1.11). For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

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.

Valid

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

Visible Damage

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

5.2 Abbreviations/Acronyms

| | |
|-----------------|--|
| °F | Degrees Fahrenheit |
| ° | Degrees |
| AC | Alternating Current |
| AOP | Abnormal Operating Procedure |
| ATWS | Anticipated Transient Without Scram |
| CDE | Committed Dose Equivalent |
| CFR | Code of Federal Regulations |
| CSFST | Critical Safety Function Status Tree |
| DBA | Design Basis Accident |
| DC | Direct Current |
| EAL | Emergency Action Level |
| EC | Emergency Coordinator |
| ECCS | Emergency Core Cooling System |
| ECL | Emergency Classification Level |
| EOF | Emergency Operations Facility |
| EOP | Emergency Operating Procedure |
| EPA | Environmental Protection Agency |
| ERG | Emergency Response Guideline |
| EPIP | Emergency Plan Implementing Procedure |
| ESF | Engineered Safety Feature |
| FAA | Federal Aviation Administration |
| FBI | Federal Bureau of Investigation |
| FEMA | Federal Emergency Management Agency |
| GE | General Emergency |
| IC | Initiating Condition |
| IPEEE | Individual Plant Examination of External Events (Generic Letter 88-20) |
| ISFSI | Independent Spent Fuel Storage Installation |
| K_{eff} | Effective Neutron Multiplication Factor |
| LCO | Limiting Condition of Operation |
| LER | Licensee Event Report |
| LOCA | Loss of Coolant Accident |
| LWR | Light Water Reactor |
| MPC | Maximum Permissible Concentration/Multi-Purpose Canister |

MSIV..... Main Steam Isolation Valve
 MSL Main Steam Line
 mR, mRem, mrem, mREM milli-Roentgen Equivalent Man
 MW Megawatt
 RCS..... Reactor Coolant System
 NEI Nuclear Energy Institute
 NESP..... National Environmental Studies Project
 NPP Nuclear Power Plant
 NRC..... Nuclear Regulatory Commission
 NSSS..... Nuclear Steam Supply System
 NORAD..... North American Aerospace Defense Command
 (NO)UE..... Notification of Unusual Event
 OBE Operating Basis Earthquake
 OCA..... Owner Controlled Area
 ODCM..... Off-site Dose Calculation Manual
 ORO Offsite Response Organization
 PA..... Protected Area
 PAG..... Protective Action Guideline
 PRA/PSA..... Probabilistic Risk Assessment / Probabilistic Safety Assessment
 PWR..... Pressurized Water Reactor
 PSIG..... Pounds per Square Inch Gauge
 R..... Roentgen
 Rem, rem, REM Roentgen Equivalent Man
 RETS..... Radiological Effluent Technical Specifications
 RNP Robinson Nuclear Plant
 RPS Reactor Protection System
 RV Reactor Vessel
 RVLIS..... Reactor Vessel Level Indicating System
 SAR Safety Analysis Report
 SBGTS Stand-By Gas Treatment System
 SBO..... Station Blackout
 SCBA..... Self-Contained Breathing Apparatus
 SG Steam Generator
 SI..... Safety Injection

SLC Selected Licensee Commitment
SPDS..... Safety Parameter Display System
SRO..... Senior Reactor Operator
SSF Safe Shutdown Facility
TEDE Total Effective Dose Equivalent
TOAF Top of Active Fuel
TSC Technical Support Center
UFSAR Updated Final Safety Analysis Report
WOG Westinghouse Owners Group

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

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

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

| RNP | NEI 99-01 Rev. 6 | |
|------------|-------------------------|--------------------|
| EAL | IC | Example EAL |
| 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 |
| FA1.1 | FA1 | 1 |
| FS1.1 | FS1 | 1 |
| FG1.1 | FG1 | 1 |
| HU1.1 | HU1 | 1, 2 3 |
| HU2.1 | HU2 | 1 |
| HU3.1 | HU3 | 1 |
| HU3.2 | HU3 | 2 |
| HU3.3 | HU3 | 3 |

| RNP | NEI 99-01 Rev. 6 | |
|------------|-------------------------|------------------------|
| EAL | IC | Example EAL |
| 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 |
| HG1.1 | HG1 | 1 |
| HG7.1 | HG7 | 1 |
| SU1.1 | SU1 | 1 |
| SU3.1 | SU2 | 1 |
| SU4.1 | SU3 | 2 |
| SU4.2 | SU3 | 1 |
| SU5.1 | SU4 | 1, 2, 3 |
| SU6.1 | SU5 | 1 |
| SU6.2 | SU5 | 2 |
| SU7.1 | SU6 | 1, 2, 3 |
| SU8.1 | SU7 | 1, 2 |
| SA1.1 | SA1 | 1 |

| RNP | NEI 99-01 Rev. 6 | |
|------------|-------------------------|------------------------|
| EAL | IC | Example EAL |
| SA3.1 | SA2 | 1 |
| SA6.1 | SA5 | 1 |
| SA9.1 | SA9 | 1 |
| SS1.1 | SS1 | 1 |
| SS2.1 | SS8 | 1 |
| SS6.1 | SS5 | 1 |
| SG1.1 | SG1 | 1 |
| SG1.2 | SG8 | 1 |
| EU1.1 | E-HU1 | 1 |

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Fission Product Barrier Matrix and Basis

Category R – Abnormal Rad Release / Rad Effluent

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

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

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

Events of this category pertain to the following subcategories:

1. Radiological Effluent

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

2. Irradiated Fuel Event

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

3. Area Radiation Levels

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

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.1 Unusual Event

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

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

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

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

| Table R-1 Effluent Monitor Classification Thresholds | | | | | | |
|--|-----------------------|------------|-------------|---------------|---------------|-------------|
| Release Point | | Monitor | GE | SAE | Alert | UE |
| Gaseous | Plant Vent | R-14C | --- | --- | --- | 2.16E+5 cpm |
| | | R-14D | 6.38E+5 cpm | 6.38E+4 cpm | 6.38E+3 cpm | --- |
| | | R-14E | 3.31E+3 cpm | 3.40E+2 cpm | 4.30E+1 cpm | --- |
| | FHB Exhaust | R-20 | --- | --- | --- | 8.06E+5 cpm |
| | FHB Exhaust HR | R-30 | --- | 2.60E+4 mR/hr | 2.60E+3 mR/hr | --- |
| Liquid | Liquid Waste Disposal | R-18 | ---- | ---- | ---- | 4.08E+6 cpm |
| | SGBD Effluent | R-19 A/B/C | ---- | ---- | ---- | 6.94E+5 cpm |
| | Condensate Polisher | R-37 | ---- | ---- | ---- | 4.23E+5 cpm |

Mode Applicability:

All

Definition(s):

None

RNP Basis:

The column "UE" gaseous and liquid release values in Table R-1 represents two times the alarm setpoint of the specified monitors. The setpoints are established to ensure the ODCM release limits are not exceeded (ref. 1, 2).

NEI 99-01 Basis:

| | | |
|----------|--------|----------------|
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|----------|--------|----------------|

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

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

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

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

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

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

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

RNP Basis Reference(s):

1. RNP-M/MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels for H. B. Robinson, Unit No. 2
2. EP-EALCALC-RNP-1401, RNP Radiological Effluent EAL Values
3. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.2 Unusual Event

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

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

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

Mode Applicability:

All

Definition(s):

None

RNP Basis:

Releases in excess of two times the site Offsite Dose Calculation Manual (ODCM) (ref. 1) instantaneous limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

NEI 99-01 Basis:

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

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

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

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

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

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

RNP Basis Reference(s):

1. H. B. Robinson Steam Electric Plant, Unit No. 2, Off-Site Dose Calculation Manual
2. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.1 Alert

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

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

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

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

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

| Table R-1 Effluent Monitor Classification Thresholds | | | | | | |
|--|-----------------------|------------|-------------|---------------|---------------|-------------|
| Release Point | | Monitor | GE | SAE | Alert | UE |
| Gaseous | Plant Vent | R-14C | --- | --- | --- | 2.16E+5 cpm |
| | | R-14D | 6.38E+5 cpm | 6.38E+4 cpm | 6.38E+3 cpm | --- |
| | | R-14E | 3.31E+3 cpm | 3.40E+2 cpm | 4.30E+1 cpm | --- |
| | FHB Exhaust | R-20 | --- | --- | --- | 8.06E+5 cpm |
| | FHB Exhaust HR | R-30 | --- | 2.60E+4 mR/hr | 2.60E+3 mR/hr | --- |
| Liquid | Liquid Waste Disposal | R-18 | ---- | ---- | ---- | 4.08E+6 cpm |
| | SGBD Effluent | R-19 A/B/C | ---- | ---- | ---- | 6.94E+5 cpm |
| | Condensate Polisher | R-37 | ---- | ---- | ---- | 4.23E+5 cpm |

Mode Applicability:

All

Definition(s):

None

RNP Basis:

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

- 10 mRem TEDE
- 50 mRem CDE Thyroid

The column "ALERT" gaseous effluent release values in Table R-1 correspond to calculated doses of 1% (10% of the calculated SAE thresholds) of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1, 2).

NEI 99-01 Basis:

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

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

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

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

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

RNP Basis Reference(s):

1. RNP-M/MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels for H. B. Robinson, Unit No. 2
2. EP-EALCALC-RNP-1401, RNP Radiological Effluent EAL Values
3. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.2 Alert

Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 3, 4)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

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

NEI 99-01 Basis:

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

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

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

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

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

RNP Basis Reference(s):

1. AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.3 Alert

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

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

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

NEI 99-01 Basis:

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

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

addresses the spectrum of possible accident events and conditions.

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

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

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

RNP Basis Reference(s):

1. H. B. Robinson Steam Electric Plant, Unit No. 2, Off-Site Dose Calculation Manual
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.4 Alert

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

EPRAD-01, Environmental Monitoring provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

NEI 99-01 Basis:

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

release).

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

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

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

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

RNP Basis Reference(s):

1. EPRAD-01, Environmental Monitoring
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.1 Site Area Emergency

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

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

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

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

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

| Table R-1 Effluent Monitor Classification Thresholds | | | | | | |
|--|-----------------------|------------|-------------|---------------|---------------|-------------|
| Release Point | | Monitor | GE | SAE | Alert | UE |
| Gaseous | Plant Vent | R-14C | --- | --- | --- | 2.16E+5 cpm |
| | | R-14D | 6.38E+5 cpm | 6.38E+4 cpm | 6.38E+3 cpm | --- |
| | | R-14E | 3.31E+3 cpm | 3.40E+2 cpm | 4.30E+1 cpm | --- |
| | FHB Exhaust | R-20 | --- | --- | --- | 8.06E+5 cpm |
| | FHB Exhaust HR | R-30 | --- | 2.60E+4 mR/hr | 2.60E+3 mR/hr | --- |
| Liquid | Liquid Waste Disposal | R-18 | ---- | ---- | ---- | 4.08E+6 cpm |
| | SGBD Effluent | R-19 A/B/C | ---- | ---- | ---- | 6.94E+5 cpm |
| | Condensate Polisher | R-37 | ---- | ---- | ---- | 4.23E+5 cpm |

Mode Applicability:

All

Definition(s):

None

RNP Basis:

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

- 100 mRem TEDE
- 500 mRem CDE Thyroid

The column "SAE" gaseous effluent release value in Table R-1 corresponds to calculated doses of 10% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1, 2).

NEI 99-01 Basis:

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

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

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

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

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

RNP Basis Reference(s):

1. RNP-M/MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels for H. B. Robinson, Unit No. 2
2. EP-EALCALC-RNP-1401, RNP Radiological Effluent EAL Values
3. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.2 Site Area Emergency

Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 3, 4)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

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

NEI 99-01 Basis:

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

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

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid

CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

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

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

RNP Basis Reference(s):

1. AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.3 Site Area Emergency

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

EPRAD-01, Environmental Monitoring provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

NEI 99-01Basis:

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

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

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

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

RNP Basis Reference(s):

1. EPRAD-01, Environmental Monitoring
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.1 General Emergency

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

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

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

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

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

| Table R-1 Effluent Monitor Classification Thresholds | | | | | | |
|--|-----------------------|------------|-------------|---------------|---------------|-------------|
| Release Point | | Monitor | GE | SAE | Alert | UE |
| Gaseous | Plant Vent | R-14C | --- | --- | --- | 2.16E+5 cpm |
| | | R-14D | 6.38E+5 cpm | 6.38E+4 cpm | 6.38E+3 cpm | --- |
| | | R-14E | 3.31E+3 cpm | 3.40E+2 cpm | 4.30E+1 cpm | --- |
| | FHB Exhaust | R-20 | --- | --- | --- | 8.06E+5 cpm |
| | FHB Exhaust HR | R-30 | --- | 2.60E+4 mR/hr | 2.60E+3 mR/hr | --- |
| Liquid | Liquid Waste Disposal | R-18 | ---- | ---- | ---- | 4.08E+6 cpm |
| | SGBD Effluent | R-19 A/B/C | ---- | ---- | ---- | 6.94E+5 cpm |
| | Condensate Polisher | R-37 | ---- | ---- | ---- | 4.23E+5 cpm |

Mode Applicability:

All

Definition(s):

None

RNP Basis:

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

- 1000 mRem TEDE
- 5000 mRem CDE Thyroid

The column “GE” gaseous effluent release values in Table R-1 correspond to calculated doses of 100% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1, 2).

NEI 99-01Basis:

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.

RNP Basis Reference(s):

1. RNP-M/MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels for H. B. Robinson, Unit No. 2
2. EP-EALCALC-RNP-1401, RNP Radiological Effluent EAL Values
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.2 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 (Notes 3, 4)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

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

NEI 99-01 Basis:

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

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

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

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

RNP Basis Reference(s):

1. AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.3 General Emergency

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

EPRAD-01, Environmental Monitoring provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

NEI 99-01 Basis:

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

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

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

RNP Basis Reference(s):

1. EPRAD-01, Environmental Monitoring
2. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Unplanned loss of water level above irradiated fuel

EAL:

RU2.1 Unusual Event

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

AND

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

- R-2 CV Area
- R-5 Spent Fuel Pit Area
- Local area survey

Mode Applicability:

All

Definition(s):

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

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

RNP Basis:

The low water level alarm in this EAL refers to the Spent Fuel Pit (SFP) low level alarm. The fuel transfer canal is normally in communication with the spent fuel pit. During refueling operations, the refueling cavity in the Containment is filled and is in communication with the fuel transfer canal when the fuel transfer tube gate valve is open. A decrease in water level in the SFP, fuel transfer canal or refueling cavity is therefore sensed by the SFP low level alarm. Neither the refueling cavity, nor the fuel transfer canal, is equipped with a low level alarm.

The specified radiation monitors are those expected to see increase area radiation levels as a result of a loss of REFUELING PATHWAY inventory (ref. 4, 5, 6). Increasing radiation indications on these monitors in the absence of indications of decreasing REFUELING CAVITY level are not classifiable under this EAL.

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

The SFP level is remotely monitored by level indicator LA-651. The level switch initiates high and low level annunciators. The Spent Fuel Pit Low Level alarm (APP-036-B6) actuates if SFP level decreases to the 36 ft. 2.5 in. (Ref. 5). In addition, the Radiation Control personnel have cameras for Containment and fuel handling building remote monitoring providing visual indication of low pool or cavity levels.

Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment. Technical Specifications Section 3.7.12 (Ref. 2) requires at least 21 ft. of water above irradiated fuel in the spent fuel pit storage racks. Technical Specifications LCO 3.9.6 (Ref. 3) requires at least 23 ft. of water above the reactor vessel flange. During refueling, this maintains sufficient water level in the fuel transfer canal, refueling cavity, and SFP to retain iodine fission product activity in the water in the event of a fuel handling accident.

While radiation monitors (e.g., R-5 Spent Fuel Pit Area Radiation Monitor or portable survey instrument) could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication, in and of itself, of whether or not inventory is being lost. Generally, elevated radiation monitor indications need to be combined with another indicator (or personnel report) of water loss.

NEI 99-01 Basis:

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

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

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

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

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

RNP Basis Reference(s):

1. USAR Section 9.1.2, Spent Fuel Storage
2. TS Section 3.7.12
3. LCO 3.9.6, Refueling Cavity Water Level
4. AOP-036, SFP Events
5. APP-036-B6, Spent Fuel Pit Low Level
6. AOP-005, Radiation Monitoring System
7. NEI 99-01 AU2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.1 Unusual Event

Uncovery of irradiated fuel in the REFUELING PATHWAY

Mode Applicability:

All

Definition(s):

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

RNP Basis:

None.

NEI 99-01 Basis:

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

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

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

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

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

RNP Basis Reference(s):

1. AOP-013 Fuel Handling Accident
2. AOP-036 SFP Events
3. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity

AND

A high alarm on **any** of the following:

- R-2 CV Area
- R-5 Spent Fuel Pit Area
- R-11/R-12 Process Monitor CV Air and Plant Vent (when sampling CV)
- R-14 Plant Vent
- R-21 Fuel Handling Building Upper Level

Mode Applicability:

All

Definition(s):

None

RNP Basis:

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

The high alarm setpoints for the radiation monitors are set to be indicative of significant increases in area and/or airborne radiation (ref. 4).

NEI 99-01 Basis:

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

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

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

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

RNP Basis Reference(s):

1. AOP-013 Fuel Handling Accident
2. AOP-036 SFP Events
3. AOP-005 Radiation Monitoring System
4. OMM-014, Radiation Monitor Setpoints
5. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.3 Alert

Lowering of spent fuel pool level to ≤ 24 ft.

Mode Applicability:

All

Definition(s):

None

RNP Basis:

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

The SFP level instruments consist of a primary channel (LI-11442A & LI-11443A) and back-up channel (LI-11442B & LI-11443B) each spanning approximately 24 ft. (14 ft. – 38 ft. indicated). Level 2 corresponds to an indicated SFP level of 24 ft. or approximately 10 ft. above the top of the SFP racks (ref. 2).

NEI 99-01 Basis:

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

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

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

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

RNP Basis Reference(s):

1. NRC EA-12-051 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. Engineering Change EC89580
3. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RS2.1 Site Area Emergency

Lowering of spent fuel pool level to ≤ 14.75 ft.

Mode Applicability:

All

Definition(s):

None

RNP Basis:

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

The SFP level instruments consist of a primary channel (LI-11442A & LI-11443A) and back-up channel (LI-11442B & LI-11443B) each spanning approximately 24 ft. (14 ft. – 38 ft. indicated). Level 3 (top of the spent fuel racks) corresponds to an SFP level of 14 ft. However, the level instruments can actually only measure to 14.75 ft. (ref. 2).

NEI 99-01 Basis:

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

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

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

RNP Basis Reference(s):

1. NRC EA-12-051 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. Engineering Change EC89580
3. NEI 99-01 AS2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RG2.1 General Emergency

Spent fuel pool level cannot be restored to at least 14.75 ft. for ≥ 60 min.
(Note 1)

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

Mode Applicability:

All

Definition(s):

None

RNP Basis:

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

The SFP level instruments consist of a primary channel (LI-11442A & LI-11443A) and back-up channel (LI-11442B & LI-11443B) each spanning approximately 24 ft. (14 ft. – 38 ft. indicated). Level 3 (top of the spent fuel racks) corresponds to an SFP level of 14 ft. However, the level instruments can actually only measure to 14.75 ft (ref. 2).

NEI 99-01 Basis:

This EAL addresses a significant loss of spent fuel pool inventory control and makeup capability leading to a prolonged uncover 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.

RNP Basis Reference(s):

1. NRC EA-12-051 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation

- 2. Engineering Change EC89580
- 3. NEI 99-01 AG2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

RA3.1 Alert

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

Control Room (R-1)

OR

Central Alarm Station (by survey)

Mode Applicability:

All

Definition(s):

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

RNP Basis:

Areas that meet this threshold include the Control Room and the Central Alarm Station (CAS). Area radiation monitor (ARM) R-1 monitors the Control room for area radiation (ref. 1, 2). The CAS is included in this EAL because of its' importance to permitting access to areas required to assure safe plant operations.

There is no permanently installed CAS area radiation monitors that may be used to assess this EAL threshold. Therefore this threshold must be assessed via local radiation survey for the CAS.

NEI 99-01 Basis:

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

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

RNP Basis Reference(s):

1. OMM-014, Radiation Monitor Setpoints
2. DBD-SD-19 Radiation Monitoring System
3. NEI 99-01 AA3

ATTACHMENT 1 EAL Bases

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

RA3.2 Alert

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

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

| Table R-2/H-2 Safe Operation & Shutdown Rooms/Areas | |
|---|----------------|
| Room/Area | Mode(s) |
| Reactor Auxiliary Building, 1 st level hallway | 1,2,3,4,5 |
| Reactor Auxiliary Building, 2 nd level hallway | 1,2,3,4,5 |
| Charging Pump Room | 1,2,3,4,5 |
| Component Cooling Water Pump Room | 1,2,3,4,5 |
| Primary Sample Room | 1,2,3,4,5 |
| Primary Demineralizer Room | 1,2,3 |
| Spent Fuel Pump / Heat Exchanger Room | 1,2,3,4,5 |
| Pipe Alley | 4 |
| RHR Heat Exchanger Room | 4 |
| RHR Pump Room entry area (access to RHR Pump CCW flow indication / control) | 4 |
| Boric Acid Batch Tank Room | 1,2,3,4,5 |
| Emergency Bus E1/E2 Room | 3,4,5 |
| Turbine Building 1 st Floor (includes Condensate Polisher, Makeup Water Treatment and Secondary Sample Room) | 1,2,3,4 |
| Turbine Building 2 nd Floor | 1,2,3,4 |
| Turbine Building 3 rd Floor | 1,3,4 |
| Containment Building | 3 |

Mode Applicability:

All

Definition(s):

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EAL Bases

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.

RNP Basis:

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

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

NEI 99-01 Basis:

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

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

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

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the elevated radiation levels). For example, the plant is in Mode 1 when the radiation increase occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.

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EAL Bases

- The increased radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

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

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

RNP Basis Reference(s):

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

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EAL Bases

Category C – Cold Shutdown / Refueling System Malfunction

EAL Group: Cold Conditions (RCS temperature $\leq 200^{\circ}\text{F}$); EALs in this category are applicable only in one or more cold operating modes.

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

The events of this category pertain to the following subcategories:

1. RCS Level

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

2. Loss of Emergency AC Power

Loss of Emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite power sources for 480V emergency buses.

3. RCS Temperature

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

4. Loss of Vital DC Power

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

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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.

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EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CU1.1 Unusual Event

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

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

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

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

RNP Basis:

RCS water level less than a required lower limit is meant to be less than the lower end of the level control band being procedurally maintained for the current condition or evolution.

With the plant in Cold Shutdown, RCS water level is normally maintained above the pressurizer low level setpoint (ref. 1, 3). However, if RCS level is being controlled below the pressurizer low level setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

With the plant in Refueling mode, RCS water level is normally maintained at or above the reactor vessel flange (ref. 2, 3, 4, 5).

NEI 99-01 Basis:

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

Refueling evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required

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EAL Bases

limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

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

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

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

RNP Basis Reference(s):

1. APP- 003 RCS & Makeup Systems
2. GP-001 Fill and Vent of the Reactor Coolant System
3. GP-008 Draining the Reactor Coolant System
4. GP-009-3 Draining the Refueling Cavity With Fuel in the Reactor Vessel
5. GP-009-5 Adjusting Reactor Vessel Level After Refueling Cavity Drain With Fuel In the Reactor
6. NEI 99-01 CU1

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CU1.2 Unusual Event

RCS water level cannot be monitored

AND EITHER

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

| Table C-1 Sumps / Tanks |
|---|
| <ul style="list-style-type: none">• Containment (CV) sump• PRT• RCDT• CCW surge tank |

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

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

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

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RNP Basis:

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

In this EAL, all water level indication is unavailable and the RCS inventory loss must be detected by indirect leakage indications. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Sumps and tanks where RCS leakage may accumulate are listed in Table C-1. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2, 3, 4).

NEI 99-01 Basis:

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

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

This EAL addresses a condition where all means to determine 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. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

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

RNP Basis Reference(s):

1. GP-001 Fill and Vent of the Reactor Coolant System
2. GP-008 Draining the Reactor Coolant System
3. GP-009-3 Draining the Refueling Cavity With Fuel in the Reactor Vessel
4. GP-009-5 Adjusting Reactor Vessel Level After Refueling Cavity Drain With Fuel In the Reactor
5. NEI 99-01 CU1

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EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory

EAL:

CA1.1 Alert

Loss of RCS inventory as indicated by RCS water level < -72 in. (69% RVLIS Full Range)

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

None

RNP Basis:

When reactor vessel water level decreases to < -72 in. (< 69% RVLIS Full Range) (ref. 1, 2), RHR pumps must be tripped. If level is below -72 inches, vortexing and air entrainment may result in damage to the RHR Pumps.

NEI 99-01 Basis:

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

For this EAL, a lowering of RCS water level below -72 inches indicates that operator actions have not been successful in restoring and maintaining RCS water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncover.

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

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

RNP Basis Reference(s):

1. GP-008, Draining the Reactor Coolant System
2. AOP-020 Loss of Residual Heat Removal (Shutdown Cooling)

ATTACHMENT 1
EAL Bases

3. NEI 99-01 CA1

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory

EAL:

CA1.2 Alert

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

AND EITHER

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

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

| Table C-1 Sumps / Tanks |
|---|
| <ul style="list-style-type: none">• Containment (CV) sump• PRT• RCDT• CCW surge tank |

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

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

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

RNP Basis:

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

ATTACHMENT 1

EAL Bases

In this EAL, all RCS water level indication would be unavailable for greater than 15 minutes, and the RCS inventory loss must be detected by indirect leakage indications. Sump level increases must be evaluated against other potential sources of leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Sumps and tanks where RCS leakage may accumulate are listed in listed in Table C-1. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2, 3, 4).

NEI 99-01 Basis:

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

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

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

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

RNP Basis Reference(s):

1. GP-001 Fill and Vent of the Reactor Coolant System
2. GP-008 Draining the Reactor Coolant System
3. GP-009-3 Draining the Refueling Cavity With Fuel in the Reactor Vessel
4. GP-009-5 Adjusting Reactor Vessel Level After Refueling Cavity Drain With Fuel In the Reactor
5. NEI 99-01 CA1

ATTACHMENT 1

EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

| |
|----------------------------------|
| CS1.1 Site Area Emergency |
|----------------------------------|

| |
|---|
| With CONTAINMENT CLOSURE not established, RCS level < 64.5% RVLIS Full Range |
|---|

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. In accordance with OMM-033, Implementation of CV Closure, Containment closure exits when:

- The equipment hatch is closed and held in place by a sufficient number of bolts as identified in CM-603, AND
- At least one door in the personnel air lock is closed, AND
- Each penetration providing direct access from the Containment atmosphere to the outside atmosphere is either:
 - closed by a manual or automatic isolation valve, blind flange, or equivalent,
 - OR
 - capable of being closed by an OPERABLE Containment Ventilation Isolation System.

RNP Basis:

64.5% RVLIS Full Range corresponds to the level of six inches below the bottom ID of the RCS hot leg penetration (240' 7" el.) (ref. 1, 2).

Six inches below the elevation of the bottom of the RCS hot leg penetration can be monitored only by RVLIS full range (64.5%). Level monitoring instruments LI-403, LI-404, Standpipe Loop B, and Standpipe Loop C cannot sense level changes in the reactor vessel below the elevation of the RCS loop hot leg penetration. The RVLIS full range threshold has been determined as follows (ref. 2, 3):

ATTACHMENT 1 EAL Bases

| Component Dimensions | | RVLIS Full Range (%) |
|---|----------|----------------------|
| Reactor Vessel bottom head OD to top of Control Rod Mechanism housing (in.) | 498.000 | NA |
| Thickness of bottom head (in.) | 5.187 | NA |
| Thickness of vessel head (in.) | 7.750 | NA |
| Height of Control Rod Mechanism above vessel closure head (in.) | 18.000 | NA |
| Inner height of vessel (in.): $498.000 - 5.187 - 7.750 - 18.000 =$ | 467.063 | 100.0 |
| Bottom of vessel (in.) | 0.000 | 0.0 |
| RVLIS span %/in.: $(100.0 - 0.0)/(467.063 - 0.000) =$ | 0.214 | NA |
| Height of RCS hot leg centerline above vessel bottom (in.) | 321.7813 | NA |
| RCS hot leg penetration diameter (in.) | 29.000 | NA |
| Height of bottom of RCS hot leg above vessel bottom (in.): $321.783 - (29.000/2) =$ | 307.2813 | A |
| 6 in. below height of bottom of hot leg (in.): $307.2813 - 6 =$ | 301.2813 | B |
| Height of top of fuel above vessel bottom (in.) | 279.5313 | C |

$$A = \text{Height of bottom of RCS hot leg above vessel bottom} \times \text{RVLIS span} = 65.8\%$$

$$B = 6 \text{ in. below height of bottom of hot leg} \times \text{RVLIS span} = 64.5\%$$

$$C = \text{Height of top of fuel above vessel bottom} \times \text{RVLIS span} = 59.8\%$$

At RNP RVLIS is normally not available when in Mode 6. The RVLIS connection to the reactor vessel head is removed prior to removing the reactor vessel head. Under those conditions where RCS level cannot be monitored, classification should be made based on CS1.3.

NEI 99-01 Basis:

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

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

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS levels of 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

ATTACHMENT 1
EAL Bases

and Low-Power Operation at Commercial Nuclear Power Plants in the United States; and NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management.

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

RNP Basis Reference(s):

1. GP-008, Draining the Reactor Coolant System
2. UFSAR Table 5.3.0-1
3. UFSAR Figure 5.3.0-1
4. NEI 99-01 CS1

ATTACHMENT 1

EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

| |
|----------------------------------|
| CS1.2 Site Area Emergency |
|----------------------------------|

| |
|--|
| With CONTAINMENT CLOSURE established, RCS level < 59.8% RVLIS Full Range |
|--|

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. In accordance with OMM-033, Implementation of CV Closure, Containment closure exits when:

- The equipment hatch is closed and held in place by a sufficient number of bolts as identified in CM-603, AND
- At least one door in the personnel air lock is closed, AND
- Each penetration providing direct access from the Containment atmosphere to the outside atmosphere is either:
 - closed by a manual or automatic isolation valve, blind flange, or equivalent,
 - OR
 - capable of being closed by an OPERABLE Containment Ventilation Isolation System.

RNP Basis:

59.8% RVLIS Full Range corresponds to the top of active fuel. Other RCS level instruments are off-scale low when core uncover occurs (ref. 1, 2).

When reactor vessel water level drops below the RVLIS full range setpoint of 59.8%, core uncover is about to occur. The RVLIS full range threshold has been determined as follows (ref. 2, 3):

ATTACHMENT 1 EAL Bases

| Component Dimensions | | RVLIS Full Range (%) |
|---|----------|----------------------|
| Reactor Vessel bottom head OD to top of Control Rod Mechanism housing (in.) | 498.000 | NA |
| Thickness of bottom head (in.) | 5.187 | NA |
| Thickness of vessel head (in.) | 7.750 | NA |
| Height of Control Rod Mechanism above vessel closure head (in.) | 18.000 | NA |
| Inner height of vessel (in.): $498.000 - 5.187 - 7.750 - 18.000 =$ | 467.063 | 100.0 |
| Bottom of vessel (in.) | 0.000 | 0.0 |
| RVLIS span %/in.: $(100.0 - 0.0)/(467.063 - 0.000) =$ | 0.214 | NA |
| Height of RCS hot leg centerline above vessel bottom (in.) | 321.7813 | NA |
| RCS hot leg penetration diameter (in.) | 29.000 | NA |
| Height of bottom of RCS hot leg above vessel bottom (in.): $321.783 - (29.000/2) =$ | 307.2813 | A |
| 6 in. below height of bottom of hot leg (in.): $307.2813 - 6 =$ | 301.2813 | B |
| Height of top of fuel above vessel bottom (in.) | 279.5313 | C |

$$A = \text{Height of bottom of RCS hot leg above vessel bottom} \times \text{RVLIS span} = 65.8\%$$

$$B = 6 \text{ in. below height of bottom of hot leg} \times \text{RVLIS span} = 64.5\%$$

$$C = \text{Height of top of fuel above vessel bottom} \times \text{RVLIS span} = 59.8\%$$

At RNP RVLIS is normally not available when in Mode 6. The RVLIS connection to the reactor vessel head is removed prior to removing the reactor vessel head. Under those conditions where RCS level cannot be monitored, classification should be made based on CS1.3.

NEI 99-01 Basis:

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

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

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS levels of 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

ATTACHMENT 1
EAL Bases

and Low-Power Operation at Commercial Nuclear Power Plants in the United States; and NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management.

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

RNP Basis Reference(s):

1. GP-008, Draining the Reactor Coolant System
2. UFSAR Table 5.3.0-1
3. UFSAR Figure 5.3.0-1
4. NEI 99-01 CS1

ATTACHMENT 1

EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.3 Site Area Emergency

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

AND

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

- UNPLANNED increase in **any** Table C-1 sump or tank due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage
- Containment High Range Radiation Monitor R-32A or R-32B > 5 R/hr
- Erratic source range monitor indication

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

| Table C-1 Sumps / Tanks |
|---|
| <ul style="list-style-type: none"> • Containment (CV) sumps • PRT • RCDT • CCW surge tank |

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

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

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

RNP Basis:

ATTACHMENT 1

EAL Bases

In this EAL, all RCS water level indication would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications. Sump level increases must be evaluated against other potential sources of leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Sumps and tanks where RCS leakage may accumulate are listed in Table C-1. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2, 3).

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed area radiation monitors (R-32A or R-32B). Any positive reading on R-32A or R-32B should be considered an indication of core uncover, either due to the shine from the uncovered core, or the initiation of clad damage. Given that the minimum range of the instrument is 1 R/hr and the instrument range is seven decades, 5 R/hr represents the lowest reading that is considered a clear positive response. If these radiation monitors reach and exceed 5 R/hr, a loss of inventory with potential to uncover the core is likely to have occurred (ref. 6).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations (ref. 4, 5).

NEI 99-01 Basis:

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

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

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

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

ATTACHMENT 1

EAL Bases

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

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

RNP Basis Reference(s):

1. GP-008, Draining the Reactor Coolant System
2. UFSAR Table 5.3.0-1
3. UFSAR Figure 5.3.0-1
4. UFSAR Section 7.2.1.1.7 Nuclear Instrumentation System
5. OP-002, Nuclear Instrumentation System
6. RNP-M/MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels for H. B. Robinson, Unit No. 2
7. NEI 99-01 CS1

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CG1.1 General Emergency

RCS level < 59.8% RVLIS Full Range for ≥ 30 min. (Note 1)

AND

Any Containment Challenge indication, Table C-2

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

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

| Table C-2 Containment Challenge Indications |
|---|
| <ul style="list-style-type: none">• CONTAINMENT CLOSURE not established (Note 6)• Containment hydrogen concentration $\geq 4\%$• UNPLANNED rise in Containment pressure |

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. In accordance with OMM-033, Implementation of CV Closure, Containment closure exits when:

- The equipment hatch is closed and held in place by a sufficient number of bolts as identified in CM-603, AND
- At least one door in the personnel air lock is closed, AND
- Each penetration providing direct access from the Containment atmosphere to the outside atmosphere is either:
 - closed by a manual or automatic isolation valve, blind flange, or equivalent,

ATTACHMENT 1 EAL Bases

OR

- capable of being closed by an OPERABLE Containment Ventilation Isolation System.

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.

RNP Basis:

59.8% RVLIS Full Range corresponds to the top of active fuel. Other RCS level instruments are off-scale low when core uncover occurs (ref. 1, 2).

The RVLIS full range threshold has been determined as follows (ref. 2, 3):

| Component Dimensions | | RVLIS Full Range (%) |
|---|----------|----------------------|
| Reactor Vessel bottom head OD to top of Control Rod Mechanism housing (in.) | 498.000 | NA |
| Thickness of bottom head (in.) | 5.187 | NA |
| Thickness of vessel head (in.) | 7.750 | NA |
| Height of Control Rod Mechanism above vessel closure head (in.) | 18.000 | NA |
| Inner height of vessel (in.): $498.000 - 5.187 - 7.750 - 18.000 =$ | 467.063 | 100.0 |
| Bottom of vessel (in.) | 0.000 | 0.0 |
| RVLIS span %/in.: $(100.0 - 0.0)/(467.063 - 0.000) =$ | 0.214 | NA |
| Height of RCS hot leg centerline above vessel bottom (in.) | 321.7813 | NA |
| RCS hot leg penetration diameter (in.) | 29.000 | NA |
| Height of bottom of RCS hot leg above vessel bottom (in.): $321.783 - (29.000/2) =$ | 307.2813 | A |
| 6 in. below height of bottom of hot leg (in.): $307.2813 - 6 =$ | 301.2813 | B |
| Height of top of fuel above vessel bottom (in.) | 279.5313 | C |

A = Height of bottom of RCS hot leg above vessel bottom x RVLIS span = 65.8%

B = 6 in. below height of bottom of hot leg x RVLIS span = 64.5%

C = Height of top of fuel above vessel bottom x RVLIS span = 59.8%

At RNP RVLIS is normally not available when in Mode 6. The RVLIS connection to the reactor vessel head is removed prior to removing the reactor vessel head. Under those conditions where RCS level cannot be monitored, classification should be made based on CG1.2.

Three conditions are associated with a challenge to containment integrity:

- CONTAINMENT CLOSURE is not established.
- In the early stages of a core uncover event, it is unlikely that hydrogen buildup due to a core uncover could result in an explosive mixture of dissolved gases in the containment. However, containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. An explosive mixture can be formed when hydrogen gas

ATTACHMENT 1

EAL Bases

concentration in the containment atmosphere is greater than or equal to 4% by volume in the presence of oxygen. Two Containment hydrogen concentration monitors (with a range of 0 to 10% hydrogen) are provided on the Core Cooling and Containment Monitor in the Control Room. Hydrogen concentration is also displayed on ERFIS Points SSC-2512A and SSC-2513A.

- Any unplanned increase in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of containment closure capability. Unplanned containment pressure increases indicates containment closure cannot be assured and the containment cannot be relied upon as a barrier to fission product release.

NEI 99-01 Basis:

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

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

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

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

In the early stages of a core uncover event, it is unlikely that hydrogen buildup due to a core uncover 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 uncover 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.

ATTACHMENT 1

EAL Bases

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*.

RNP Basis Reference(s):

1. GP-008, Draining the Reactor Coolant System
2. UFSAR Table 5.3.0-1
3. UFSAR Figure 5.3.0-1
4. NEI 99-01 CG1

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CG1.2 General Emergency

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

AND

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

- UNPLANNED increase in **any** Table C-1 sump or tank due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage
- Containment High Range Radiation Monitor R-32A or R-32B > 5 R/hr
- Erratic source range monitor indication

AND

Any Containment Challenge indication, Table C-2

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

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

| Table C-1 Sumps / Tanks |
|---|
| <ul style="list-style-type: none">• Containment sumps• PRT• RCDT• CCW surge tank |

ATTACHMENT 1
EAL Bases

| Table C-2 Containment Challenge Indications |
|---|
| <ul style="list-style-type: none">• CONTAINMENT CLOSURE not established (Note 6)• Containment hydrogen concentration $\geq 4\%$• UNPLANNED rise in Containment pressure |

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. In accordance with OMM-033, Implementation of CV Closure, Containment closure exits when:

- The equipment hatch is closed and held in place by a sufficient number of bolts as identified in CM-603, AND
- At least one door in the personnel air lock is closed, AND
- Each penetration providing direct access from the Containment atmosphere to the outside atmosphere is either:
 - closed by a manual or automatic isolation valve, blind flange, or equivalent,
 - OR
 - capable of being closed by an OPERABLE Containment Ventilation Isolation System.

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.

RNP Basis:

In this EAL, all RCS water level indication would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications. Sump level increases must be evaluated against other potential sources of leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Sumps and tanks where RCS leakage may accumulate are listed in listed in Table C-1. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed area radiation monitors (R-32A or R-32B). Any positive reading on R-32A or R-32B should be considered an indication of core uncover, either due to the shine from the uncovered core, or

ATTACHMENT 1

EAL Bases

the initiation of clad damage. Given that the minimum range of the instrument is 1 R/hr and the instrument range is seven decades, 5 Rem/hr represents the lowest reading that is considered a clear positive response. If these radiation monitors reach and exceed 5 R/hr, a loss of inventory with potential to uncover the core is likely to have occurred (ref. 6).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations (ref. 4, 5).

Three conditions are associated with a challenge to containment integrity:

- CONTAINMENT CLOSURE is not established.
- In the early stages of a core uncover event, it is unlikely that hydrogen buildup due to a core uncover could result in an explosive mixture of dissolved gases in the containment. However, containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. An explosive mixture can be formed when hydrogen gas concentration in the containment atmosphere is greater than 4% by volume in the presence of oxygen.
- Any unplanned increase in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of containment closure capability. Unplanned containment pressure increases indicates containment closure cannot be assured and the containment cannot be relied upon as a barrier to fission product release.

NEI 99-01 Basis:

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

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

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

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

In the early stages of a core uncover event, it is unlikely that hydrogen buildup due to a core uncover could result in an explosive gas mixture in containment. If all installed hydrogen gas

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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 uncover has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

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

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

RNP Basis Reference(s):

1. GP-008, Draining the Reactor Coolant System
2. UFSAR Table 5.3.0-1
3. UFSAR Figure 5.3.0-1
4. UFSAR Section 7.2.1.1.7 Nuclear Instrumentation System
5. OP-002, Nuclear Instrumentation System
6. RNP-M/MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels for H. B. Robinson, Unit No. 2
7. NEI 99-01 CG1

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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CU2.1 Unusual Event

AC power capability to 480V emergency buses E-1 and E-2 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 Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

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

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.

RNP Basis:

Emergency buses E-1 and E-2 are the essential buses.

For emergency classification purposes, “capability” means that an AC power source is available to the emergency buses, whether or not the buses are powered from it.

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Offsite power is connected to the 230KV switchyard through six lines:

- Darlington County Plant south through 230 KV CB 52/11 and 230 KV CB 52/12
- Darlington County Plant north through 230 KV CB 52/13 and 230 KV CB 52/14
- Sumter through 230 KV CB 52/10 and 230 KV CB 52/11
- Rockingham through 230 KV CB 52/2 and 230 KV CB 52/3
- Darlington SCPSA through 230 KV CB 52/7 and 230 KV CB 52/8
- Florence through 230 KV CB 52/4 and 230 KV CB 52/5

The Unit 1 115KV switchyard is connected to the 230KV through two lines:

- #1 Auto transformer to 115 KV west bus through 230 KV CBs 52/5 and 52/6
- #2 Auto transformer to 115 KV east bus through 230 KV CBs 52/1 and 52/2

Emergency bus E-1 is supplied from 4160V bus 2 through Station Service Transformer 2F (SST 2F). When the Main Generator is shutdown, 4160V bus 2 is supplied from the Startup Transformer. Emergency bus E-2 is supplied from 4160V bus 3 through Station Service Transformer 2G (SST 2G). 4160V bus 3 is normally supplied from the Startup Transformer. The first source of offsite emergency power is the 115 KV to 4160V Startup Transformer. This transformer is supplied from the lines connecting to the 115KV grid and two ties to the 230KV grid.

Another method to obtain offsite power is by backfeeding the emergency buses through the Main and Unit Auxiliary Transformer. This is only done during Cold Shutdown unless nuclear safety considerations require it to be done during Hot Shutdown when no other power sources are available. Buses E-1 and E-2 can also be supplied from onsite Emergency Diesel Generators (EDGs) A and B, respectively (ref.1, 2, 3, 4, 5, 6).

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

NEI 99-01 Basis:

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

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

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

- A loss of all offsite power with a concurrent failure of all but one emergency power

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source (e.g., an onsite diesel generator).

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

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

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

RNP Basis Reference(s):

1. OP-603, Electrical Distribution
2. UFSAR Figure 8.1.2-1a
3. UFSAR Section 8.3 Onsite Power Systems
4. AOP-024, Loss of Instrument Bus
5. DBD/R87038/SD16, Electrical Distribution System
6. EOP-ECA-0.0, Loss of All AC Power
7. NEI 99-01 CU2

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EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CA2.1 Alert

Loss of **all** offsite and **all** onsite AC power capability to 480V emergency buses E-1 and E-2 for ≥ 15 min. (Note 1)

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

Mode Applicability:

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

Definition(s):

None

RNP Basis:

Emergency buses E-1 and E-2 are the essential buses.

For emergency classification purposes, “capability” means that an AC power source is available to the emergency buses, whether or not the buses are powered from it.

Offsite power is connected to the 230KV switchyard through six lines:

- Darlington County Plant south through 230 KV CB 52/11 and 230 KV CB 52/12
- Darlington County Plant north through 230 KV CB 52/13 and 230 KV CB 52/14
- Sumter through 230 KV CB 52/10 and 230 KV CB 52/11
- Rockingham through 230 KV CB 52/2 and 230 KV CB 52/3
- Darlington SCPSA through 230 KV CB 52/7 and 230 KV CB 52/8
- Florence through 230 KV CB 52/4 and 230 KV CB 52/5

The Unit 1 115KV switchyard is connected to the 230KV through two lines:

- #1 Auto transformer to 115 KV west bus through 230 KV CBs 52/5 and 52/6
- #2 Auto transformer to 115 KV east bus through 230 KV CBs 52/1 and 52/2

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Emergency bus E-1 is supplied from 4160V bus 2 through Station Service Transformer 2F (SST 2F). When the Main Generator is shutdown, 4160V bus 2 is supplied from the Startup Transformer. Emergency bus E-2 is supplied from 4160V bus 3 through Station Service Transformer 2G (SST 2G). 4160V bus 3 is normally supplied from the Startup Transformer. The first source of offsite emergency power is the 115 KV to 4160V Startup Transformer. This transformer is supplied from the lines connecting to the 115KV grid and two ties to the 230KV grid

Another method to obtain offsite power is by backfeeding the emergency buses through the Main and Unit Auxiliary Transformer. This is only done during Cold Shutdown unless nuclear safety considerations require it to be done during Hot Shutdown when no other power sources are available. Buses E-1 and E-2 can also be supplied from onsite Emergency Diesel Generators (EDGs) A and B, respectively (ref.1, 2, 3, 4, 5, 6).

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

NEI 99-01 Basis:

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

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

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

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

RNP Basis Reference(s):

1. OP-603, Electrical Distribution
2. UFSAR Figure 8.1.2-1a
3. UFSAR Section 8.3 Onsite Power Systems
4. AOP-024, Loss of Instrument Bus
5. DBD/R87038/SD16, Electrical Distribution System
6. EOP-ECA-0.0, Loss of All AC Power
7. NEI 99-01 CA2

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EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.1 Unusual Event

UNPLANNED increase in RCS temperature to > 200°F due to loss of decay heat removal capability

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

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.

RNP Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specifications cold shutdown temperature limit (200°F, ref. 1). These include Core Exit Thermocouples (CETs) and the wide range (0-700°F) RTDs located in the hot and cold legs of the RCS:

| | <u>Cold Leg</u> | <u>Hot Leg</u> |
|--------|-----------------|-----------------------|
| Loop 1 | TE-410 | TE-413-1 and TE-413-2 |
| Loop 2 | TE-420 | TE-423 |
| Loop 3 | TE-430 | TE-433 |

TE-413 is a dual element RTD with TE-413-1 providing indication to TR-413, and TE-413-2 providing indication to the Inadequate Core Cooling Monitor (ICCM). Temperatures are also recorded on TR-413 (T_{hot}) and TR-410 (T_{cold}), which are located on the RTGB. The temperatures of the hot and cold legs can also be read on the core subcooling monitor (designated T_h 1, T_h 2, T_h 3, T_c 1, T_c 2, and T_c 3) and are used for indication during heatup and cooldown. RCS/RHR pump discharge temperature indication, such as TR-604, can also be used to monitor RCS temperature (ref. 2, 3).

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

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EAL Bases

NEI 99-01 Basis:

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

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

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

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

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.

RNP Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. GP-002, Cold Shutdown to Hot Subcritical at No-Load T-AVG
3. GP-007, Plant Cooldown from Hot Shutdown to Cold Shutdown Conditions
4. NEI 99-01 CU3

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EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

| |
|----------------------------|
| CU3.2 Unusual Event |
|----------------------------|

| |
|---|
| Loss of all RCS temperature and RCS level indication for ≥ 15 min. (Note 1) |
|---|

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

Mode Applicability:

5 - Cold Shutdown, 6- Refueling

Definition(s):

None

RNP Basis:

Reactor vessel water level is normally monitored using the following instruments (ref. 2):

- LI-403
- LI-404
- Standpipe Loop B
- Standpipe Loop C
- RVLIS (LT-511AB and LT-511BB)
- Remote camera, if vessel head is removed

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specifications cold shutdown temperature limit of 200°F (ref. 1). These include Core Exit Thermocouples (CETs) and the wide range (0-700°F) RTDs located in the hot and cold legs of the RCS:

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| <u>Cold Leg</u> | <u>Hot Leg</u> | |
|-----------------|----------------|-----------------------|
| Loop 1 | TE-410 | TE-413-1 and TE-413-2 |
| Loop 2 | TE-420 | TE-423 |
| Loop 3 | TE-430 | TE-433 |

TE-413 is a dual element RTD with TE-413-1 providing indication to TR-413, and TE-413-2 providing indication to the Inadequate Core Cooling Monitor (ICCM). Temperatures are also recorded on TR-413 (T_{hot}) and TR-410 (T_{cold}), which are located on the RTGB. The temperatures of the hot and cold legs can also be read on the core subcooling monitor (designated T_h 1, T_h 2, T_h 3, T_c 1, T_c 2, and T_c 3) and are used for indication during heatup and cooldown. RCS/RHR pump discharge temperature indication, such as TR-604, can also be used to monitor RCS temperature (ref. 3, 4).

NEI 99-01 Basis:

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

RNP Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. GP-008, Draining the Reactor Coolant System
3. GP-002, Cold Shutdown to Hot Subcritical at No-Load T-AVG
4. GP-007, Plant Cooldown from Hot Shutdown to Cold Shutdown Conditions
5. NEI 99-01 CU3

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EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.1 Alert

UNPLANNED increase in RCS temperature to > 200°F for > Table C-3 duration
(Note 1)

OR

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

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

| Table C-3: RCS Heat-up Duration Thresholds | | |
|--|----------------------------|------------------|
| RCS Status | Containment Closure Status | Heat-up Duration |
| Intact (but not REDUCED INVENTORY) | N/A | 60 min.* |
| Not intact OR At REDUCED INVENTORY | established | 20 min.* |
| | not established | 0 min. |
| * If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable. | | |

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. In accordance with OMM-033, Implementation of CV Closure, Containment closure exits when:

- The equipment hatch is closed and held in place by a sufficient number of bolts as identified in CM-603, AND
- At least one door in the personnel air lock is closed, AND
- Each penetration providing direct access from the Containment atmosphere to the outside atmosphere is either:

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- closed by a manual or automatic isolation valve, blind flange, or equivalent,
OR
- capable of being closed by an OPERABLE Containment Ventilation Isolation System.

REDUCED INVENTORY - Plant condition when fuel is in the Reactor Vessel and Reactor Coolant System level is less than or equal to -36 inches below the vessel flange.

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.

RNP Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specifications cold shutdown temperature limit of 200°F (ref. 1). These include the wide range (0-700°F) RTDs located in the hot and cold legs of the RCS:

| <u>Cold Leg</u> | <u>Hot Leg</u> | |
|-----------------|----------------|-----------------------|
| Loop 1 | TE-410 | TE-413-1 and TE-413-2 |
| Loop 2 | TE-420 | TE-423 |
| Loop 3 | TE-430 | TE-433 |

TE-413 is a dual element RTD with TE-413-1 providing indication to TR-413, and TE-413-2 providing indication to the Inadequate Core Cooling Monitor (ICCM). Temperatures are also recorded on TR-413 (T_{hot}) and TR-410 (T_{cold}), which are located on the RTGB. The temperatures of the hot and cold legs can also be read on the core subcooling monitor (designated $T_h 1$, $T_h 2$, $T_h 3$, $T_c 1$, $T_c 2$, and $T_c 3$) and are used for indication during heatup and cooldown. RCS/RHR pump discharge temperature indication, such as TR-604, can also be used to monitor RCS temperature (ref. 2, 3).

PI-403, RCS Narrow Range Pressure (0-1000 psi), is graduated in 20 psi increments and is capable of measuring pressure to less than 10 psig (ref. 2).

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

ATTACHMENT 1

EAL Bases

NEI 99-01 Basis:

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

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

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

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

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

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

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

RNP Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. GP-002, Cold Shutdown to Hot Subcritical at No-Load T-AVG
3. GP-007, Plant Cooldown from Hot Shutdown to Cold Shutdown Conditions
6. NEI 99-01 CA3

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EAL Bases

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

< 109.5 VDC (Bus A) / < 106.2 (Bus B) bus voltage indications on Technical Specification required 125 VDC buses for ≥ 15 min. (Note 1)

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

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

None

RNP Basis:

The A and B batteries are safety-related and are equipped with two redundant battery chargers per bus. The A and B batteries are sized to carry expected shutdown loads following a design basis accident with no battery chargers available for a period of 1 hour without battery terminal voltage falling below minimum allowable voltage. The four safety-related chargers are sized to charge a partially discharged battery within 24 hours while carrying its normal load (ref. 1).

Minimum battery terminal voltage is 1.75 VDC per cell for each of 60 cells per battery or 105 VDC (ref. 1). Calculations performed for the B battery replacement, however, specify minimum battery terminal voltage of 106.8 VDC and a corresponding bus voltage of 106.2 VDC (ref. 2). Battery A minimum bus voltage was calculated to be 109.5 VDC (ref. 11). Control Room annunciator APP-036-D3, BATT A/B LO VOLT, is received at 123 VDC and signals sustained loss of a battery charger or battery/cell failure (ref. 3). Battery bus voltage is indicated on ERFIS Points APV3022A (MCC-A) and APV3023A (MCC-B) (ref. 4).

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

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EAL Bases

NEI 99-01 Basis

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

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

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

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

RNP Basis Reference(s):

1. UFSAR Section 8.3.2 DC Power System (125 Volt)
2. Calculation RNP-E-6.031, Station Battery B Replacement, Figure 1
3. APP-036-D3 BATT A/B LO VOLT
4. UFSAR Figure 8.3.1-5
5. Technical Specifications 3.8.4, DC Sources - Operating
6. Technical Specifications 3.8.5, DC Sources - Shutdown
7. Technical Specifications 3.8.6, Battery Cell Parameters
8. OP-601, DC Supply System
9. EPP-26, Loss of DC Bus A
10. EPP-27, Loss of DC Bus B
11. Calculation RNP-E-6.018, Section 5.1.4
12. NEI 99-01 CU4

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EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.1 Unusual Event

Loss of **all** Table C-4 onsite communication methods

OR

Loss of **all** Table C-4 offsite communication methods

OR

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

| Table C-4 Communication Methods | | | |
|--|---------------|----------------|------------|
| System | Onsite | Offsite | NRC |
| Public Address System | X | | |
| PBX Telephone System | X | | |
| Radio Transceivers for RNP and Vicinity | X | | |
| Back-up Telephone System (ESSX) | X | | |
| Plant Security Radio Transceivers | X | | |
| Corporate Telephone Communications System (Voicenet) | | X | X |
| BellSouth | | X | X |
| Dedicated Telephone System to Load Dispatcher | | X | |
| Plant Security Radio Control Station | | X | |
| DEMNET | | X | |
| NRC Emergency Telecommunication System (ETS) | | | X |
| Satellite Phones | | X | X |
| Cellular Phones | | X | X |
| Palmetto 800 Transceivers | | X | |

Mode Applicability:

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

Definition(s):

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None

RNP Basis:

Onsite, offsite and NRC communications include one or more of the systems listed in Table C-4 (ref. 1, 2).

The NRC ETS Phone and the NRC HPN Phone are part of the PABX and will be unavailable if the PABX is unavailable.

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

NEI 99-01 Basis:

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

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

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

The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the State, Darlington, Lee and Chesterfield County EOCs

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

RNP Basis Reference(s):

1. PLP-007, Emergency Plan, Attachment 6.1
2. UFSAR Section 9.5.2 Communications Systems
3. NEI 99-01 CU5

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EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting Safety Systems

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

EAL:

CA6.1 Alert

The occurrence of **any** Table C-5 hazardous event

AND EITHER:

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

| Table C-5 Hazardous Events |
|--|
| <ul style="list-style-type: none">• 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:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

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

ATTACHMENT 1

EAL Bases

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

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

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

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

- (1) The integrity of the reactor coolant pressure boundary;
- (2) The capability to shut down the reactor and maintain it in a safe shutdown condition;
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

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

RNP Basis:

- The significance of seismic events are discussed under EAL HU2.1 (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2).
- The plant Seismic Category I structures are designed to withstand the effects of the design wind, 83 mph (108 gust). (ref. 3, 4).
- An explosion that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

ATTACHMENT 1
EAL Bases

NEI 99-01 Basis:

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

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

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

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

RNP Basis Reference(s):

1. AOP-021 Seismic Disturbances
2. RNP-F/PSA-0009, Assessment of Internal Flooding Events
3. UFSAR Table 3.3.1-1
4. OMM-021, Operation During Adverse Weather Conditions
5. NEI 99-01 CA6

ATTACHMENT 1
EAL Bases

Category H – Hazards and Other Conditions Affecting Plant Safety

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

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

1. Security

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

2. Seismic Event

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

3. Natural or Technology Hazard

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

4. Fire

Fires can pose significant hazards to personnel and reactor safety. Appropriate for classification are fires within the site Protected Area or 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.

ATTACHMENT 1

EAL Bases

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

ATTACHMENT 1

EAL Bases

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1 Unusual Event

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

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

HOSTILE ACTION - An act toward RNP 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 RNP. 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).

RNP Basis:

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

Reports from Security Shift Supervision may be made via non-supervisory security personnel such as the CAS operator.

ATTACHMENT 1

EAL Bases

NEI 99-01 Basis:

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

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1, 2). 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 RNP Security Plan and DBT (ref. 1).

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 the RNP Security Plan and DBT (ref. 1).

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

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

RNP Basis Reference(s):

1. RNP Security Plan and DBT
2. AOP-034, Security Events
3. NEI 99-01 HU1

ATTACHMENT 1

EAL Bases

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Hostile action within the owner controlled area or airborne attack threat

EAL:

HA1.1 Alert

A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the 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):

HOSTILE ACTION - An act toward RNP 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 RNP. 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 - That area surrounding the Protected Area beyond which RNP exercises access control.

RNP Basis:

Reports from Security Shift Supervision may be made via non-supervisory security personal such as the CAS operator.

NEI 99-01 Basis:

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

Timely and accurate communications between the Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1, 2).

ATTACHMENT 1

EAL Bases

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*.

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

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

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

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

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

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

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

RNP Basis Reference(s):

1. RNP Security Plan and DBT
2. AOP-034, Security Events
3. NEI 99-01 HA1

ATTACHMENT 1

EAL Bases

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Hostile Action within the Protected Area

EAL:

HS1.1 Site Area Emergency

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

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated Security area around the process buildings and is depicted in Drawing HBR2-9800, Plot Plan RNP.

RNP Basis:

Reports from Security Shift Supervision may be made via non-supervisory security personal such as the CAS operator.

NEI 99-01 Basis:

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

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1, 2).

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*.

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

ATTACHMENT 1

EAL Bases

sheltering). The Site Area Emergency declaration will mobilize Offsite Response Organization (ORO) resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

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

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

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

RNP Basis Reference(s):

1. RNP Security Plan and DBT
2. AOP-034, Security Events
3. NEI 99-01 HS1

ATTACHMENT 1

EAL Bases

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HG1.1 General Emergency

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

AND EITHER of the following has occurred:

Any of the following safety functions cannot be controlled or maintained

- Reactivity
- Core cooling
- RCS heat removal

OR

Damage to spent fuel has occurred or is IMMINENT

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward RNP 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 RNP. 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

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated Security area around the process buildings and is depicted in Drawing HBR2-9800, Plot Plan RNP.

ATTACHMENT 1
EAL Bases

RNP Basis:

Reports from Security Shift Supervision may be made via non-supervisory security personnel such as the CAS operator.

NEI 99-01 Basis:

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

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1, 2).

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*.

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 RNP Security Plan and DBT (ref.1).

RNP Basis Reference(s):

1. RNP Security Plan and DBT
2. AOP-034, Security Events
3. NEI 99-01 HG1

ATTACHMENT 1

EAL Bases

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 Recording Unit A or B indicates seismic event > Operating Basis Earthquake (0.1g horizontal **OR** 0.067g vertical)

Mode Applicability:

All

Definition(s):

None

RNP Basis:

AOP-021 Seismic Disturbances provides the guidance for determining if the OBE earthquake threshold is exceeded and any required response actions (ref. 1).

The Operating Basis Earthquake (OBE) is defined as that earthquake which could reasonably be expected to affect the plant site during the operating life of the plant, based on the earthquake potential of the geographic area. At Robinson Plant, this is defined as half of the vibration defined for an SSE or 0.1g horizontal or 0.067g vertical. Facility design ensures that all equipment necessary to operate the plant without undue risk to the health and safety of the public will remain functional for any seismic event where ground motion is less than that of the OBE (ref. 1, 2).

If the OBE ALARM on Seismic Monitor "B" is illuminated, then the earthquake is greater than an Operating Basis Earthquake (0.1g Horizontal or 0.067g Vertical). If the DBE/SSE ALARM on Seismic Monitor "A" is illuminated, then the earthquake is greater than the Design Basis/Safe Shutdown Earthquake (0.2g Horizontal OR 0.133g Vertical) (ref. 1).

To avoid inappropriate emergency classification resulting from spurious actuation of the seismic instrumentation or felt motion not attributable to seismic activity, an offsite agency (USGS, National Earthquake Information Center) can confirm that an earthquake has occurred in the area of the plant. Such confirmation should not, however, preclude a timely emergency declaration based on receipt of the OBE alarm. When calling the NEIC, select **option #1** and inform the analyst you wish to confirm recent seismic activity in the vicinity of RNP. Provide the analyst with the following RNP coordinates: **34° 24.2' north latitude, 80° 09.5' west longitude** (ref. 3). Alternatively, near real-time seismic activity can be accessed via the NEIC website:

ATTACHMENT 1

EAL Bases

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

NEI 99-01 Basis:

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

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

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

RNP Basis Reference(s):

1. AOP-21 Seismic Disturbances
2. UFSAR 3.7.4 Seismic Instrumentation
3. UFSAR 1.2.1 Site and Environment
4. NEI 99-01 HU2

ATTACHMENT 1

EAL Bases

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 - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated Security area around the process buildings and is depicted in Drawing HBR2-9800, Plot Plan RNP.

RNP Basis:

Response actions associated with a tornado onsite are provided in OMM-021, Operation During Adverse Weather Conditions (ref. 1).

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

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

NEI 99-01 Basis:

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

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

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

RNP Basis Reference(s):

1. OMM-021, Operation During Adverse Weather Conditions
2. NEI 99-01 HU3

ATTACHMENT 1
EAL Bases

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 needed for the current operating mode

Mode Applicability:

All

Definition(s):

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

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

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

- (1) The integrity of the reactor coolant pressure boundary;
- (2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition;
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

RNP Basis:

The primary internal flooding area of concern is the Auxiliary Building (includes the Component Cooling Water Pump Room). Flooding in this area could have the potential to cause a reactor trip and could result in consequential failures to important systems. The potential for flooding in this area was determined by an examination of piping systems in the area and also considered propagation of water from one area to another. The most important internal flooding initiating events are associated with the failure of large service water pipes located on elevation 226 in the Auxiliary Building. Elevation 226 is mostly open to flood propagation, and significant safety-related equipment could be affected by a very large flood that would allow water to accumulate to significant depths. When water level reaches a certain height in the Auxiliary Building, both trains of safeguards equipment can be rendered inoperable. This event

ATTACHMENT 1

EAL Bases

is further compounded by the fact that all spilled water may become contaminated and must be treated so until proven otherwise. Other flood initiating events on elevation 226 are less important but do have the potential to affect safe plant operations if timely mitigation does not occur to terminate flooding (Ref. 1, 2).

NEI 99-01 Basis:

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

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

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

RNP Basis Reference(s):

1. RNP-F/PSA-0009, Assessment of Internal Flooding Events
2. RSC 99-17, RNP Probabilistic Safety Assessment, Section 3.5, Internal Flooding Initiating Events, Table 3.12
3. AOP-022, Loss of Service Water
4. AOP-032, Response to Flooding from the Fire Protection System
5. NEI 99-01 HU3

ATTACHMENT 1

EAL Bases

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 offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated Security area around the process buildings and is depicted in Drawing HBR2-9800, Plot Plan RNP.

RNP Basis:

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

NEI 99-01 Basis:

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

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

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

RNP Basis Reference(s):

1. NEI 99-01 HU3

ATTACHMENT 1

EAL Bases

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):

None

RNP Basis:

None

NEI 99-01 Basis:

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

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

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

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

RNP Basis Reference(s):

1. NEI 99-01 HU3

ATTACHMENT 1

EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.1 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 Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas

- Containment
- Auxiliary Building
- Control Room
- Fuel Handling Building
- Intake
- AFW Room
- 4 KV Switchgear Room
- E-1/E-2 Switchgear Room
- RWST
- CST

Mode Applicability:

All

ATTACHMENT 1

EAL Bases

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.

RNP Basis:

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

Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1, 2).

NEI 99-01 Basis:

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

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

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

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

RNP Basis Reference(s):

1. OMP-003, Shutdown Safety Function Guidelines
2. OMM-003, Fire Protection Pre-Plans/Unit No. 2
3. NEI 99-01 HU4

ATTACHMENT 1

EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.2 Unusual Event

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

AND

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

AND

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

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

**Table H-1
Fire Areas**

- Containment
- Auxiliary Building
- Control Room
- Fuel Handling Building
- Intake
- AFW Room
- 4 KV Switchgear Room
- E-1/E-2 Switchgear Room
- RWST
- CST

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.

ATTACHMENT 1 EAL Bases

RNP Basis:

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

Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1, 2).

NEI 99-01 Basis:

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

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

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

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

Basis-Related Requirements from Appendix R

Appendix R to 10 CFR 50, states in part:

Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."

When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.

Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the

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EAL Bases

need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in this EAL, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

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

RNP Basis Reference(s):

1. OMP-003, Shutdown Safety Function Guidelines
2. OMM-003, Fire Protection Pre-Plans/Unit No. 2
3. NEI 99-01 HU4

ATTACHMENT 1
EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.3 Unusual Event

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

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

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated Security area around the process buildings and is depicted in Drawing HBR2-9800, Plot Plan RNP.

RNP Basis:

None

NEI 99-01 Basis:

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

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

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

RNP Basis Reference(s):

1. NEI 99-01 HU4

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EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.4 Unusual Event

A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

Mode Applicability:

All

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 - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated Security area around the process buildings and is depicted in Drawing HBR2-9800, Plot Plan RNP.

RNP Basis:

None

NEI 99-01 Basis:

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

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

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

RNP Basis Reference(s):

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EAL Bases

1. NEI 99-01 HU4

ATTACHMENT 1
EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gases

Initiating Condition: Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

HA5.1 Alert

Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table R-2/H-2 rooms or areas

AND

Entry into the room or area is prohibited or IMPEDED (Note 5)

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

| Table R-2/H-2 Safe Operation & Shutdown Rooms/Areas | |
|---|----------------|
| Room/Area | Mode(s) |
| Reactor Auxiliary Building, 1 st level hallway | 1,2,3,4,5 |
| Reactor Auxiliary Building, 2 nd level hallway | 1,2,3,4,5 |
| Charging Pump Room | 1,2,3,4,5 |
| Component Cooling Water Pump Room | 1,2,3,4,5 |
| Primary Sample Room | 1,2,3,4,5 |
| Primary Demineralizer Room | 1,2,3 |
| Spent Fuel Pump / Heat Exchanger Room | 1,2,3,4,5 |
| Pipe Alley | 4 |
| RHR Heat Exchanger Room | 4 |
| RHR Pump Room entry area (access to RHR Pump CCW flow indication / control) | 4 |
| Boric Acid Batch Tank Room | 1,2,3,4,5 |
| Emergency Bus E1/E2 Room | 3,4,5 |
| Turbine Building 1 st Floor (includes Condensate Polisher, Makeup Water Treatment and Secondary Sample Room) | 1,2,3,4 |
| Turbine Building 2 nd Floor | 1,2,3,4 |
| Turbine Building 3 rd Floor | 1,3,4 |
| Containment Building | 3 |

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EAL Bases

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).

RNP Basis:

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

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

A list of hazardous gases (ref. 2, 3):

| Flammable Gas (1) or Toxic (2) | Asphyxiant Gas |
|---------------------------------------|--|
| Acetylene (1, 2) | Nitrogen |
| Oxygen (1) | Argon |
| Propane (1, 2) | Carbon Dioxide |
| Hydrogen (1) | Halon |
| Ammonia (1, 2) | Helium Freon – Genetron Dichlorodifluormethane |
| P-10 Gas, used in portal monitors (1) | Freon – R-22, Chlorodifluoromethane |
| Ethanolamine (1, 2) | |
| Methoxypropylamine (1, 2) | |
| Dimethylamine (1, 2) | |

NEI 99-01 Basis:

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

An Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous

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EAL Bases

release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.

Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the Emergency Coordinator's judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

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

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.
- 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 automatically or manually activate a fire suppression system in an area..

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

RNP Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases
2. PLP-021, Chemical Storage, Inventory, Spill and Hazard Communication Program

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3. PLP-022, Environmental Regulatory Compliance Guidelines for Disposal of Hazardous Waste/Surplus Chemicals
4. NEI 99-01 HA5

ATTACHMENT 1

EAL Bases

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 Dedicated/Alternate Shutdown System

Mode Applicability:

All

Definition(s):

None

RNP Basis:

The Shift Manager (SM) determines if the Control Room is uninhabitable and requires evacuation. Control Room uninhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (ref. 1, 2).

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

NEI 99-01 Basis:

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

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

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

ATTACHMENT 1
EAL Bases

RNP Basis Reference(s):

1. AOP-004, Control Room Inaccessibility
2. DSP-002, Hot Shutdown Using the Dedicated/Alternate Shutdown System
3. NEI 99-01 HA6

ATTACHMENT 1

EAL Bases

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 Dedicated/Alternate Shutdown System

AND

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

- Reactivity
- Core Cooling
- RCS heat removal

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

Mode Applicability:

All

Definition(s):

None

RNP Basis:

The Shift Manager determines if the Control Room is uninhabitable and requires evacuation. Control Room inhabitation may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (ref. 1, 2).

NEI 99-01 Basis:

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

The determination of whether or not “control” is established at the remote safe shutdown location(s) is based on Emergency Coordinator judgment. The Emergency Coordinator 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).

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EAL Bases

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

RNP Basis Reference(s):

1. AOP-004, Control Room Inaccessibility
2. DSP-002, Hot Shutdown Using the Dedicated/Alternate Shutdown System
3. NEI 99-01 HS6

ATTACHMENT 1

EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a UE

EAL:

HU7.1 Unusual Event

Other conditions exist which in the judgment of the Emergency Coordinator 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.

RNP Basis:

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the RNP Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage

ATTACHMENT 1
EAL Bases

the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

NEI 99-01 Basis:

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

RNP Basis Reference(s):

1. PLP-007, Robinson Emergency Plan
2. NEI 99-01 HU7

ATTACHMENT 1
EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions exist that in the judgment of the Emergency Coordinator warrant declaration of an Alert

EAL:

HA7.1 Alert

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

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward RNP 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 RNP. 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).

RNP Basis:

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the RNP Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref.1).

NEI 99-01 Basis:

ATTACHMENT 1
EAL Bases

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 Coordinator to fall under the emergency classification level description for an Alert.

RNP Basis Reference(s):

1. PLP-007, Robinson Emergency Plan
2. NEI 99-01 HA7

ATTACHMENT 1
EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a Site Area Emergency

EAL:

HS7.1 Site Area Emergency

Other conditions exist which in the judgment of the Emergency Coordinator 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 RNP 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 RNP. 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).

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the RNP Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage

ATTACHMENT 1
EAL Bases

the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

NEI 99-01 Basis:

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

RNP Basis Reference(s):

1. PLP-007, Robinson Emergency Plan
2. NEI 99-01 HS7

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EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

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

EAL:

HG7.1 General Emergency

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

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward RNP 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 RNP. 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.

RNP Basis:

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the RNP Emergency Response Plan. The Operations Shift Manager(SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

ATTACHMENT 1
EAL Bases

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

NEI 99-01 Basis:

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

RNP Basis Reference(s):

1. PLP-007, Robinson Emergency Plan
2. NEI 99-01 HG7

ATTACHMENT 1 EAL Bases

Category S – System Malfunction

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

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

The events of this category pertain to the following subcategories:

1. Loss of Emergency AC Power

Loss of emergency electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite sources for 480V emergency buses.

2. Loss of Vital DC Power

Loss of emergency electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of vital plant 125 VDC power sources.

3. Loss of Control Room Indications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant increase from these base-line levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

5. RCS Leakage

The reactor vessel provides a volume for the coolant that covers the reactor core. The reactor 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

ATTACHMENT 1

EAL Bases

cracks that may propagate to an extent threatening fuel clad, RCS and containment integrity.

6. RPS Failure

This subcategory includes events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RPS to complete a reactor trip comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean any trip failure event that does not achieve reactor shutdown (< 5% reactor power). If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and containment integrity.

7. Loss of Communications

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

8. Containment Failure

Failure of containment isolation capability (under conditions in which the containment is not currently challenged) or loss of containment depressurization system capability warrants emergency classification.

9. Hazardous Event Affecting Safety Systems

Various natural and technological events that result in degraded plant safety system performance or significant visible damage warrant emergency classification under this subcategory.

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EAL Bases

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

| | |
|--------------|----------------------|
| SU1.1 | Unusual Event |
|--------------|----------------------|

| | |
|--|--|
| | Loss of all offsite AC power capability to 480V emergency buses E-1 and E-2 for ≥ 15 min. (Note 1) |
|--|--|

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

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

RNP Basis:

Emergency buses E-1 and E-2 are the essential buses.

For emergency classification purposes, “capability” means that an AC power source is available to the emergency buses, whether or not the buses are powered from it.

Offsite power is connected to the 230KV switchyard through six lines:

- Darlington County Plant south through 230 KV CB 52/11 and 230 KV CB 52/12
- Darlington County Plant north through 230 KV CB 52/13 and 230 KV CB 52/14
- Sumter through 230 KV CB 52/10 and 230 KV CB 52/11
- Rockingham through 230 KV CB 52/2 and 230 KV CB 52/3
- Darlington SCPSA through 230 KV CB 52/7 and 230 KV CB 52/8
- Florence through 230 KV CB 52/4 and 230 KV CB 52/5

The Unit 1 115KV switchyard is connected to the 230KV through two lines:

- #1 Auto transformer to 115 KV west bus through 230 KV CBs 52/5 and 52/6
- #2 Auto transformer to 115 KV east bus through 230 KV CBs 52/1 and 52/2

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EAL Bases

Emergency bus E-1 is supplied from 4160V bus 2 through Station Service Transformer 2F (SST 2F). When the Main Generator is in service, 4160V bus 2 is supplied from the Unit Auxiliary Transformer through 4160V bus 1 and tie breaker 52/10. When the Main Generator is shutdown, 4160V bus 2 is supplied from the Startup Transformer. Emergency bus E-2 is supplied from 4160V bus 3 through Station Service Transformer 2G (SST 2G). 4160V bus 3 is normally supplied from the Startup Transformer. The Startup Transformer is supplied from the lines connecting to the 115KV grid and two ties to the 230KV grid.

Another method to obtain offsite power is by backfeeding the emergency buses through the Main and Unit Auxiliary Transformer. This is only done during Cold Shutdown unless nuclear safety considerations require it to be done during Hot Shutdown when no other power sources are available. Buses E-1 and E-2 can also be supplied from onsite Emergency Diesel Generators (EDGs) A and B, respectively (ref.1, 2, 3, 4, 5, 6).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses.

NEI 99-01 Basis:

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC emergency buses. This condition represents a potential reduction in the level of safety of the plant.

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

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

Escalation of the emergency classification level would be via IC SA1.

RNP Basis Reference(s):

1. OP-603, Electrical Distribution
2. UFSAR Figure 8.1.2-1a
3. UFSAR Section 8.3 Onsite Power Systems
4. AOP-024, Loss of Instrument Bus
5. DBD/R87038/SD16, Electrical Distribution System
6. EOP-ECA-0.0, Loss of All AC Power
7. NEI 99-01 SU1

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SA1.1 Alert

AC power capability to 480V emergency buses E-1 and E-2 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 Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

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

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

- (1) The integrity of the reactor coolant pressure boundary;
- (2) The capability to shut down the reactor and maintain it in a safe shutdown condition;
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

RNP Basis:

Emergency buses E-1 and E-2 are the essential buses.

For emergency classification purposes, “capability” means that an AC power source is available to the emergency buses, whether or not the buses are powered from it.

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Offsite power is connected to the 230KV switchyard through six lines:

- Darlington County Plant south through 230 KV CB 52/11 and 230 KV CB 52/12
- Darlington County Plant north through 230 KV CB 52/13 and 230 KV CB 52/14
- Sumter through 230 KV CB 52/10 and 230 KV CB 52/11
- Rockingham through 230 KV CB 52/2 and 230 KV CB 52/3
- Darlington SCPSA through 230 KV CB 52/7 and 230 KV CB 52/8
- Florence through 230 KV CB 52/4 and 230 KV CB 52/5

The Unit 1 115KV switchyard is connected to the 230KV through two lines:

- #1 Auto transformer to 115 KV west bus through 230 KV CBs 52/5 and 52/6
- #2 Auto transformer to 115 KV east bus through 230 KV CBs 52/1 and 52/2

Emergency bus E-1 is supplied from 4160V bus 2 through Station Service Transformer 2F (SST 2F). When the Main Generator is in service, 4160V bus 2 is supplied from the Unit Auxiliary Transformer through 4160V bus 1 and tie breaker 52/10. When the Main Generator is shutdown, 4160V bus 2 is supplied from the Startup Transformer. Emergency bus E-2 is supplied from 4160V bus 3 through Station Service Transformer 2G (SST 2G). 4160V bus 3 is normally supplied from the Startup Transformer. The Startup Transformer is supplied from the lines connecting to the 115KV grid and two ties to the 230KV grid.

Another method to obtain offsite power is by backfeeding the emergency buses through the Main and Unit Auxiliary Transformer. This is only done during Cold Shutdown unless nuclear safety considerations require it to be done during Hot Shutdown when no other power sources are available. Buses E-1 and E-2 can also be supplied from onsite Emergency Diesel Generators (EDGs) A and B, respectively (ref.1, 2, 3, 4, 5, 6).

NEI 99-01 Basis:

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

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

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

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

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

Escalation of the emergency classification level would be via IC SS1.

RNP Basis Reference(s):

1. OP-603, Electrical Distribution
2. UFSAR Figure 8.1.2-1a
3. UFSAR Section 8.3 Onsite Power Systems
4. AOP-024, Loss of Instrument Bus
5. DBD/R87038/SD16, Electrical Distribution System
6. EOP-ECA-0.0, Loss of All AC Power
7. NEI 99-01 SA1

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SS1.1 Site Area Emergency

Loss of **all** offsite and **all** onsite AC power capability to 480V emergency buses E-1 and E-2 for ≥ 15 min. (Note 1)

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

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

Emergency buses E-1 and E-2 are the essential buses.

This EAL is indicated by the loss of all offsite and onsite AC power capability to 480V emergency buses E-1 and E-2. For emergency classification purposes, “capability” means that an AC power source is available to the emergency buses, whether or not the buses are powered from it.

Offsite power is connected to the 230KV switchyard through six lines:

- Darlington County Plant south through 230 KV CB 52/11 and 230 KV CB 52/12
- Darlington County Plant north through 230 KV CB 52/13 and 230 KV CB 52/14
- Sumter through 230 KV CB 52/10 and 230 KV CB 52/11
- Rockingham through 230 KV CB 52/2 and 230 KV CB 52/3
- Darlington SCPSA through 230 KV CB 52/7 and 230 KV CB 52/8
- Florence through 230 KV CB 52/4 and 230 KV CB 52/5

The Unit 1 115KV switchyard is connected to the 230KV through two lines:

- #1 Auto transformer to 115 KV west bus through 230 KV CBs 52/5 and 52/6

ATTACHMENT 1

EAL Bases

- #2 Auto transformer to 115 KV east bus through 230 KV CBs 52/1 and 52/2

Emergency bus E-1 is supplied from 4160V bus 2 through Station Service Transformer 2F (SST 2F). When the Main Generator is in service, 4160V bus 2 is supplied from the Unit Auxiliary Transformer through 4160V bus 1 and tie breaker 52/10. When the Main Generator is shutdown, 4160V bus 2 is supplied from the Startup Transformer. Emergency bus E-2 is supplied from 4160V bus 3 through Station Service Transformer 2G (SST 2G). 4160V bus 3 is normally supplied from the Startup Transformer. The Startup Transformer is supplied from the lines connecting to the 115KV grid and two ties to the 230KV grid.

Another method to obtain offsite power is by backfeeding the emergency buses through the Main and Unit Auxiliary Transformer. This is only done during Cold Shutdown unless nuclear safety considerations require it to be done during Hot Shutdown when no other power sources are available. Buses E-1 and E-2 can also be supplied from onsite Emergency Diesel Generators (EDGs) A and B, respectively (ref.1, 2, 3, 4, 5, 6).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. The interval begins when both offsite and onsite AC power capability are lost.

NEI 99-01 Basis:

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

RNP Basis Reference(s):

1. OP-603, Electrical Distribution
2. UFSAR Figure 8.1.2-1a
3. UFSAR Section 8.3 Onsite Power Systems
4. AOP-024, Loss of Instrument Bus
5. DBD/R87038/SD16, Electrical Distribution System
6. EOP-ECA-0.0, Loss of All AC Power
7. NEI 99-01 SS1

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EAL Bases

Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power capability to 480V emergency buses E-1 and E-2

AND EITHER:

- Restoration of at least one emergency bus in < 8 hours is **not** likely (Note 1)
- Core Cooling **RED** Path entry conditions met

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

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

This EAL is indicated by the extended loss of all offsite and onsite AC power capability 480V emergency buses E-1 and E-2 either for greater than the RNP Station Blackout (SBO) coping analysis time (8 hrs.) (ref. 7) or that has resulted in indications of an actual loss of adequate core cooling.

Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED Path conditions being met. (ref. 8).

Offsite power is connected to the 230KV switchyard through six lines:

- Darlington County Plant south through 230 KV CB 52/11 and 230 KV CB 52/12
- Darlington County Plant north through 230 KV CB 52/13 and 230 KV CB 52/14
- Sumter through 230 KV CB 52/10 and 230 KV CB 52/11
- Rockingham through 230 KV CB 52/2 and 230 KV CB 52/3
- Darlington SCPSA through 230 KV CB 52/7 and 230 KV CB 52/8

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- Florence through 230 KV CB 52/4 and 230 KV CB 52/5

The Unit 1 115KV switchyard is connected to the 230KV through two lines:

- #1 Auto transformer to 115 KV west bus through 230 KV CBs 52/5 and 52/6
- #2 Auto transformer to 115 KV east bus through 230 KV CBs 52/1 and 52/2

Emergency bus E-1 is supplied from 4160V bus 2 through Station Service Transformer 2F (SST 2F). When the Main Generator is in service, 4160V bus 2 is supplied from the Unit Auxiliary Transformer through 4160V bus 1 and tie breaker 52/10. When the Main Generator is shutdown, 4160V bus 2 is supplied from the Startup Transformer. Emergency bus E-2 is supplied from 4160V bus 3 through Station Service Transformer 2G (SST 2G). 4160V bus 3 is normally supplied from the Startup Transformer. The Startup Transformer is supplied from the lines connecting to the 115KV grid and two ties to the 230KV grid.

Another method to obtain offsite power is by backfeeding the emergency buses through the Main and Unit Auxiliary Transformer. This is only done during Cold Shutdown unless nuclear safety considerations require it to be done during Hot Shutdown when no other power sources are available. Buses E-1 and E-2 can also be supplied from onsite Emergency Diesel Generators (EDGs) A and B, respectively (ref. 1, 2, 3, 4, 5, 6).

Eight hours is the station blackout coping time (ref 7).

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on Emergency Coordinator judgment as it relates to imminent Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers. Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED Path conditions being met (ref. 8). Specifically, Core Cooling RED Path conditions exist if either core exit T/Cs are reading greater than or equal to 1200°F or subcooling is less than 18°F [37°F] AND no RCPs are running AND core exit T/Cs are reading greater than or equal to 700°F AND RVLIS Full Range is less than 41% (ref. 8).

NEI 99-01 Basis:

This IC addresses a prolonged loss of all power sources to AC emergency buses. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A prolonged loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

The EAL should require declaration of a General Emergency prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC emergency bus by the end of the analyzed station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of challenges to multiple fission product barriers.

ATTACHMENT 1

EAL Bases

The estimate for restoring at least one emergency bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, protective actions for the public.

The EAL will also require a General Emergency declaration if the loss of AC power results in parameters that indicate an inability to adequately remove decay heat from the core.

RNP Basis Reference(s):

1. OP-603, Electrical Distribution
2. UFSAR Figure 8.1.2-1a
3. UFSAR Section 8.3 Onsite Power Systems
4. AOP-024, Loss of Instrument Bus
5. DBD/R87038/SD16, Electrical Distribution System
6. EOP-ECA-0.0, Loss of All AC Power
7. 8S19-P-101, H.B. Robinson, Unit No. 2 Station Blackout Coping Analysis Report
8. Critical Safety Function Status Trees, CSF-2 Core Cooling
9. NEI 99-01 SG1

ATTACHMENT 1
EAL Bases

Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** 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 480V emergency buses E-1 and E-2 for ≥ 15 min.

AND

Loss of **all** vital DC power based on < 109.5 VDC Bus A and < 106.2 VDC Bus B voltage indications for ≥ 15 min.

(Note 1)

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

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

This EAL is indicated by the loss of all offsite and onsite emergency AC power capability to 480V emergency buses E-1 and E-2 for greater than 15 minutes in combination with degraded vital DC power voltage. This EAL addresses operating experience from the March 2011 accident at Fukushima Daiichi.

Offsite power is connected to the 230KV switchyard through six lines:

- Darlington County Plant south through 230 KV CB 52/11 and 230 KV CB 52/12
- Darlington County Plant north through 230 KV CB 52/13 and 230 KV CB 52/14
- Sumter through 230 KV CB 52/10 and 230 KV CB 52/11
- Rockingham through 230 KV CB 52/2 and 230 KV CB 52/3
- Darlington SCPSA through 230 KV CB 52/7 and 230 KV CB 52/8
- Florence through 230 KV CB 52/4 and 230 KV CB 52/5

The Unit 1 115KV switchyard is connected to the 230KV through two lines:

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EAL Bases

- #1 Auto transformer to 115 KV west bus through 230 KV CBs 52/5 and 52/6
- #2 Auto transformer to 115 KV east bus through 230 KV CBs 52/1 and 52/2

Emergency bus E-1 is supplied from 4160V bus 2 through Station Service Transformer 2F (SST 2F). When the Main Generator is in service, 4160V bus 2 is supplied from the Unit Auxiliary Transformer through 4160V bus 1 and tie breaker 52/10. When the Main Generator is shutdown, 4160V bus 2 is supplied from the Startup Transformer. Emergency bus E-2 is supplied from 4160V bus 3 through Station Service Transformer 2G (SST 2G). 4160V bus 3 is normally supplied from the Startup Transformer. The Startup Transformer is supplied from the lines connecting to the 115KV grid and two ties to the 230KV grid.

Another method to obtain offsite power is by backfeeding the emergency buses through the Main and Unit Auxiliary Transformer. This is only done during Cold Shutdown unless nuclear safety considerations require it to be done during Hot Shutdown when no other power sources are available. Buses E-1 and E-2 can also be supplied from onsite Emergency Diesel Generators (EDGs) A and B, respectively (ref.1, 2, 3, 4, 5, 6).

The A and B batteries are safety-related and are equipped with two redundant battery chargers per bus. The A and B batteries are sized to carry expected shutdown loads following a design basis accident with no battery chargers available for a period of 1 hour without battery terminal voltage falling below minimum allowable voltage. The four safety-related chargers are sized to charge a partially discharged battery within 24 hours while carrying its normal load.

Minimum battery terminal voltage is 1.75 VDC per cell for each of 60 cells per battery or 105 VDC (ref. 3). Calculations performed for the B battery replacement, however, specify minimum battery terminal voltage of 106.8 VDC and a corresponding bus voltage of 106.2 VDC (ref. 7). Battery A minimum bus voltage was calculated to be 109.5 VDC (ref. 8). Control Room annunciator APP-036-D3, BATT A/B LO VOLT, is received at 123 VDC and signals sustained loss of a battery charger or battery/cell failure (ref. 9). Battery bus voltage is indicated on ERFIS Points APV3022A (MCC-A) and APV3023A (MCC-B).

NEI-9901 Basis:

This IC addresses a concurrent and prolonged loss of both emergency AC and Vital DC power. A loss of all emergency AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both emergency AC and vital DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

ATTACHMENT 1
EAL Bases

RNP Basis Reference(s):

1. OP-603, Electrical Distribution
2. UFSAR Figure 8.1.2-1a
3. UFSAR Section 8.3 Onsite Power Systems
4. AOP-024, Loss of Instrument Bus
5. DBD/R87038/SD16, Electrical Distribution System
6. EOP-ECA-0.0, Loss of All AC Power
7. Calculation RNP-E-6.031, Station Battery B Replacement, Figure 1
8. Calculation RNP-E-6.018, Section 5.1.4
9. APP-036-D3, BATT A/B LO VOLT
10. NEI 99-01 SG8

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

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

EAL:

| | |
|--------------|----------------------------|
| SS2.1 | Site Area Emergency |
|--------------|----------------------------|

| | |
|--|--|
| | Loss of all vital DC power based on < 109.5 VDC Bus A and < 106.2 VDC Bus B voltage indications for ≥ 15 min. (Note 1) |
|--|--|

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

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

The A and B batteries are safety-related and are equipped with two redundant battery chargers per bus. The A and B batteries are sized to carry expected shutdown loads following a design basis accident with no battery chargers available for a period of 1 hour without battery terminal voltage falling below minimum allowable voltage. The four safety-related chargers are sized to charge a partially discharged battery within 24 hours while carrying its normal load.

Minimum battery terminal voltage is 1.75 VDC per cell for each of 60 cells per battery or 105 VDC (ref. 1). Calculations performed for the B battery replacement, however, specify minimum battery terminal voltage of 106.8 VDC and a corresponding bus voltage of 106.2 VDC (ref. 2). Battery A minimum bus voltage was calculated to be 109.5 VDC (ref. 3). Control Room annunciator APP-036-D3, BATT A/B LO VOLT, is received at 123 VDC and signals sustained loss of a battery charger or battery/cell failure (ref. 4). Battery bus voltage is indicated on ERFIS Points APV3022A (MCC-A) and APV3023A (MCC-B).

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EAL Bases

NEI 99-01 Basis:

This IC addresses a loss of vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

RNP Basis Reference(s):

1. UFSAR Section 8.3 Onsite Power Systems
2. Calculation RNP-E-6.031, Station Battery B Replacement, Figure 1
3. Calculation RNP-E-6.018, Section 5.1.4
4. APP-036-D3, BATT A/B LO VOLT
5. NEI 99-01 SS8

ATTACHMENT 1
EAL Bases

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-1 parameters from within the Control Room for ≥ 15 min. (Note 1)

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

Table S-1 Safety System Parameters

- Reactor power
- RCS level
- RCS pressure
- Core exit T/C temperature
- Level in at least one SG
- Auxiliary feed flow in at least one SG

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

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

RNP Basis:

SAFETY SYSTEM parameters listed in Table S-1 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. SPDS/ERFIS plant computer serve as a redundant compensatory indicators which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

ATTACHMENT 1

EAL Bases

NEI 99-01 Basis:

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an “inability to monitor” means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via IC SA3.

RNP Basis Reference(s):

1. AOP-025, RTGB Instrument Failure
2. AOP-024, Loss of Instrument Bus
3. NEI 99-01 SU2

ATTACHMENT 1
EAL Bases

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-1 parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

Any significant transient is in progress, Table S-2

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

Table S-1 Safety System Parameters

- Reactor power
- RCS level
- RCS pressure
- Core exit T/C temperature
- Level in at least one SG
- Auxiliary feed flow in at least one SG

Table S-2 Significant Transients

- Reactor trip
- Runback > 25% thermal power
- Electrical load rejection > 25% electrical load
- Safety injection actuation

ATTACHMENT 1

EAL Bases

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

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

RNP Basis:

SAFETY SYSTEM parameters listed in Table S-1 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. SPDS/ERFIS plant computer serve as a redundant compensatory indicators which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

Significant transients are listed in Table S-2 and include response to automatic or manually initiated functions such as reactor trips, runbacks involving greater than 25% thermal power change, electrical load rejections of greater than 25% full electrical load or SI injection actuations.

NEI 99-01 Basis:

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an “inability to monitor” means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other

ATTACHMENT 1
EAL Bases

SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via ICs FS1 or IC RS1

RNP Basis Reference(s):

1. AOP-025, RTGB Instrument Failure
2. AOP-024, Loss of Instrument Bus
3. NEI 99-01 SA2

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

| | |
|--------------|----------------------|
| SU4.1 | Unusual Event |
|--------------|----------------------|

| | |
|--|---|
| | RCS activity > Technical Specification Section 3.4.16 limits (Note 9) |
|--|---|

Note 9: Mode 3 applicable only when RCS temperature is $\geq 500^{\circ}\text{F}$.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby

Definition(s):

None

RNP Basis:

This EAL addresses reactor coolant samples exceeding Technical Specification 3.4.8 which are applicable in Modes 1, 2, 3 and 4. The Technical Specification limits accommodate an iodine spike phenomenon that may occur following changes in thermal power. The Technical Specification LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident (ref. 1).

Consistent with the Technical Specification 3.4.16 RCS activity limit applicability, this EAL is only applicable in Modes 1, 2 and Mode 3 when RCS temperature is $\geq 500^{\circ}\text{F}$ (ref. 1).

NEI 99-01 Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

RNP Basis Reference(s):

1. Technical Specification 3.4.16 RCS Specific Activity
2. NEI 99-01 SU3

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

| | |
|--------------|----------------------|
| SU4.2 | Unusual Event |
|--------------|----------------------|

| |
|---|
| With letdown in service, letdown line area radiation monitor R-9 > 500 mR/hr (Note 9) |
|---|

Note 9: Mode 3 applicable only when RCS temperature is $\geq 500^{\circ}\text{F}$.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby

Definition(s):

None

RNP Basis:

The normal CVCS charging and letdown flow path allows purification of the reactor coolant and control of the RCS volume. Downstream of the non-regenerative heat exchanger and upstream of the mixed bed demineralizers, the letdown stream passes by area radiation monitor R-9, which is mounted above the letdown line pipe. In order for R-9 readings to represent fission product activity in the reactor coolant and thereby warn of potential fuel element failure, letdown must be in service allowing flow through the letdown line and past the radiation monitor.

The threshold value of 500 mR/hr represents fuel failure in excess of 0.1% and indicates a challenge to the Technical Specifications allowable limits for fuel clad degradation (ref. 1, 2, 3).

Consistent with the Technical Specification 3.4.16 RCS activity limit applicability, this EAL is only applicable in Modes 1, 2 and Mode 3 when RCS temperature is $\geq 500^{\circ}\text{F}$ (ref. 3).

NEI 99-01 Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

RNP Basis Reference(s):

ATTACHMENT 1
EAL Bases

1. RNP-M/MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels
2. OMM-014, Radiation Monitor Setpoints
3. Technical Specification 3.4.16 RCS Specific Activity
4. NEI 99-01 SU3

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1 Unusual Event

RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min.

OR

RCS identified leakage > 25 gpm for ≥ 15 min.

OR

Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min.
(Note 1)

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

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

Water balance inventory calculations are normally used to determine RCS leakage. ERFIS Group Display SP5, RCS Leakage Paths, is used to evaluate parameters that are indicative of an RCS leakage source (ref. 1).

Technical Specifications (ref. 2, 3) defines RCS leakage as follows:

Identified Leakage:

- a. Leakage from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or return) that is captured and conducted to collection systems or a sump or collecting tank, or
- b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage, or
- c. RCS leakage through a steam generator to the Secondary Coolant System (primary-to-secondary leakage).

ATTACHMENT 1 EAL Bases

Unidentified Leakage:

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

Pressure Boundary Leakage:

Pressure boundary leakage is leakage (except SG leakage) through a nonisolable leak in an RCS component body, pipe wall, or vessel wall.

RCS leakage outside of the containment that is not considered identified or unidentified leakage per Technical Specifications includes leakage via interfacing systems such as RCS to the Component Cooling Water, or systems that directly see RCS pressure outside containment such as Chemical & Volume Control System, Safety Injection, Nuclear Sampling System and Residual Heat Removal System (when in the shutdown cooling mode) (ref. 1, 4).

The existence of leakage from the RCS to the Containment, regardless of the source of leakage, may be detected by one or more of the following conditions (ref. 5):

- The Containment air particulate monitor (R-12) is quite sensitive to low leak rates. The Containment radiogas monitor can be used as a backup to the air particulate monitor.
- A leakage detection system is included which determines leakage losses from water and steam systems within the Containment, including that from the RCS. This system collects and measures moisture condensed from the Containment atmosphere by the cooling coils of the Containment air recirculation cooling units. This system provides a means of measuring leakage, including leaks from the cooling coils themselves which are part of the Containment boundary.
- An increase in the amount of coolant makeup water which is required to maintain normal level in the pressurizer, or an increase in Containment sump level are also used as leakage detection methods.

Escalation of this EAL to the Alert level is via Category F, Fission Product Barrier Degradation, EAL FA1.1.

NEI 99-01 Basis:

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

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

ATTACHMENT 1

EAL Bases

conditions thus apply to leakage into the containment, a secondary-side system (e.g., steam generator tube leakage) or a location outside of containment.

The leak rate values for each condition were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). The first condition uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. An emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated).

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

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

RNP Basis Reference(s):

1. AOP-016, Excessive Primary Plant Leakage
2. Technical Specifications section 1.1 Definitions
3. Technical Specifications 3.4.13, RCS Operational Leakage
4. OST-051, Reactor Coolant System Leakage Evaluation
5. UFSAR Section 5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary
6. NEI 99-01 SU4

ATTACHMENT 1

EAL Bases

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

SU6.1 Unusual Event

An automatic trip did **not** shut down the reactor as indicated by reactor power $\geq 5\%$ after **any** RPS setpoint is exceeded

AND

A subsequent automatic trip or manual trip action taken at the RTGB is successful in shutting down the reactor as indicated by reactor power $< 5\%$ (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

RNP Basis:

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) trip function. A reactor trip is automatically initiated by the RPS when certain continuously monitored parameters exceed predetermined setpoints (ref. 1).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from an automatic reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful trip has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2, 3).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the RTGB (reactor trip pushbuttons). Reactor shutdown achieved by use of other trip actions specified in FRP-S.1 Response to Nuclear Power

ATTACHMENT 1

EAL Bases

Generation/ATWS (remote reactor trip breakers, generator circuit breakers, tripping rod drive motors, tripping the turbine, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 2).

Following any automatic RPS trip signal, EOP-E-0 (ref. 1) and FRP-S.1 (ref. 2) prescribe insertion of redundant manual trip signals to back up the automatic RPS trip function and ensure reactor shutdown is achieved. Even if the first subsequent manual trip signal inserts all control rods to the full-in position immediately after the initial failure of the automatic trip, the lowest level of classification that must be declared is an Unusual Event.

In the event that the operator identifies a reactor trip is imminent and initiates a successful manual reactor trip before the automatic RPS trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. However, if subsequent manual reactor trip actions fail to reduce reactor power below 5%, the event escalates to the Alert under EAL SA6.1.

If by procedure, operator actions include the initiation of an immediate manual trip following receipt of an automatic trip signal and there are no clear indications that the automatic trip failed (such as a time delay following indications that a trip setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic trip or manual actions. If a subsequent review of the trip actuation indications reveals that the automatic trip did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

NEI 99-01 Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic trip is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor trip, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip using a different switch). Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

ATTACHMENT 1

EAL Bases

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be “at the reactor control consoles”.

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor trip signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the trip failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

RNP Basis Reference(s):

1. EOP-E-0 Reactor Trip or Safety Injection
2. FRP-S.1 Response to Nuclear Power Generation/ATWS
3. CSFST CSF-1 Subcriticality
4. NEI 99-01 SU5

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

SU6.2 Unusual Event

A manual trip did **not** shut down the reactor as indicated by reactor power $\geq 5\%$ after **any** manual trip action was initiated

AND

A subsequent automatic trip or manual trip action taken at the RTGB is successful in shutting down the reactor as indicated by reactor power $< 5\%$ (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

RNP Basis:

This EAL addresses a failure of a manually initiated trip in the absence of having exceeded an automatic RPS trip setpoint and a subsequent automatic or manual trip is successful in shutting down the reactor (reactor power $< 5\%$). (ref. 2).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from a manual reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful trip has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 1, 2, 3).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the RTGB (reactor trip pushbuttons). Reactor shutdown achieved by use of other trip actions specified in FRP-S.1 Response to Nuclear Power Generation/ATWS (remote reactor trip breakers, generator circuit breakers, tripping rod drive

ATTACHMENT 1

EAL Bases

motors, tripping the turbine, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 2).

If both subsequent automatic and subsequent manual reactor trip actions in the Control Room fail to reduce reactor power below the power associated with the safety system design ($< 5\%$) following a failure of an initial manual trip, the event escalates to an Alert under EAL SA6.1

NEI 99-01 Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic trip is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor trip, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip using a different switch). Depending upon several factors, the initial or subsequent effort to manually shutdown the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

ATTACHMENT 1

EAL Bases

Should a reactor trip signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip and the RTS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the trip failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

RNP Basis Reference(s):

1. EOP-E-0 Reactor Trip or Safety Injection
2. FRP-S.1 Response to Nuclear Power Generation/ATWS
3. CSFST CSF-1 Subcriticality
4. NEI 99-01 SU5

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor

EAL:

SA6.1 Alert

An automatic or manual trip fails to shut down the reactor as indicated by reactor power $\geq 5\%$

AND

Manual trip actions taken at the RTGB are **not** successful in shutting down the reactor as indicated by reactor power $\geq 5\%$ (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

RNP Basis:

This EAL addresses any automatic or manual reactor trip signal that fails to shut down the reactor followed by a subsequent manual trip that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed.

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the RTGB (reactor trip pushbuttons). Reactor shutdown achieved by use of other trip actions specified in FRP-S.1 Response to Nuclear Power Generation/ATWS (remote reactor trip breakers, generator circuit breakers, tripping rod drive motors, tripping the turbine, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 2).

5% rated power is a minimum reading on the power range scale that indicates continued power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below 5%, plant response will be similar to that observed during a

ATTACHMENT 1

EAL Bases

normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than 5 % power (ref. 1, 2, 3).

Escalation of this event to a Site Area Emergency would be under EAL SS6.1 or Emergency Coordinator judgment.

NEI 99-01 Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control console (e.g., locally opening breakers). Actions taken at backpanels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control console".

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shut down the reactor is prolonged enough to cause a challenge to the core cooling or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6 or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

RNP Basis Reference(s):

1. EOP-E-0 Reactor Trip or Safety Injection
2. FRP-S.1 Response to Nuclear Power Generation/ATWS
3. CSFST CSF-1 Subcriticality

ATTACHMENT 1
EAL Bases

4. NEI 99-01 SA5

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual trip fails to shut down the reactor as indicated by reactor power $\geq 5\%$

AND

All actions to shut down the reactor are **not** successful as indicated by reactor power $\geq 5\%$

AND EITHER:

- Core Cooling **RED** Path entry conditions met
- Heat Sink **RED** Path entry conditions met

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

RNP Basis:

This EAL addresses the following:

- Any automatic reactor trip signal followed by a manual trip that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed (EAL SA6.1), **AND**
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS barriers.

Reactor shutdown achieved by use of FRP-S.1 Response to Nuclear Power Generation/ATWS (remote reactor trip breakers, generator circuit breakers, tripping rod drive motors, tripping the turbine, emergency boration or manually driving control rods) are also

ATTACHMENT 1

EAL Bases

credited as a successful manual trip provided reactor power can be reduced below 5% before indications of an extreme challenge to either core cooling or heat removal exist (ref. 1, 2, 3, 4, 5).

5% rated power is a minimum reading on the power range scale that indicates continued power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below 5%, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than 5% power (ref. 1, 2).

Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED Path conditions being met. Specifically, Core Cooling RED Path conditions exist if either core exit T/Cs are reading greater than or equal to 1200°F or subcooling is less than 18°F [37°F] AND no RCPs are running AND core exit T/Cs are reading greater than or equal to 700°F AND RVLIS Full Range is less than 41% (ref. 4).

Indication of inability to adequately remove heat from the RCS is manifested by CSFST Heat Sink RED Path conditions being met. Specifically, Heat Sink RED Path conditions exist if narrow range level in at least one steam generator is not greater than or equal to 9% [18%] and total feedwater flow to the intact steam generators is less than 300 gpm or 0.2E6 PPH (ref. 5).

NEI 99-01 Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shut down the reactor. The inclusion of this IC and EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Escalation of the emergency classification level would be via IC RG1 or FG1.

RNP Basis Reference(s):

1. EOP-E-0 Reactor Trip or Safety Injection
2. FRP-S.1 Response to Nuclear Power Generation/ATWS

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3. CSFST CSF-1 Subcriticality
4. CSFST CSF-2 Core Cooling
5. CSFST CSF-3 Heat Sink
6. NEI 99-01 SS5

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EAL Bases

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

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

EAL:

SU7.1 Unusual Event

Loss of **all** Table S-3 onsite communication methods

OR

Loss of **all** Table S-3 offsite communication methods

OR

Loss of **all** Table S-3 NRC communication methods

| Table S-3 Communication Methods | | | |
|--|---------------|----------------|------------|
| System | Onsite | Offsite | NRC |
| Public Address System | X | | |
| PBX Telephone System | X | | |
| Radio Transceivers for RNP and Vicinity | X | | |
| Back-up Telephone System (ESSX) | X | | |
| Plant Security Radio Transceivers | X | | |
| Corporate Telephone Communications System (Voicenet) | | X | X |
| BellSouth | | X | X |
| Dedicated Telephone System to Load Dispatcher | | X | |
| Plant Security Radio Control Station | | X | |
| DEMNET | | X | |
| NRC Emergency Telecommunication System (ETS) | | | X |
| Satellite Phones | | X | X |
| Cellular Phones | | X | X |
| Palmetto 800 Transceivers | | X | |

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

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None

RNP Basis:

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

The NRC ETS Phone and the NRC HPN Phone are part of the PABX and will be unavailable if the PABX is unavailable.

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

NEI 99-01 Basis:

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

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

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

The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the State, Darlington, Lee and Chesterfield County EOCs

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

RNP Basis Reference(s):

1. PLP-007, Emergency Plan, Attachment 6.1
2. UFSAR Section 9.5.2 Communications Systems
3. NEI 99-01 SU6

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EAL Bases

Category: S – System Malfunction

Subcategory: 8 – Containment Failure

Initiating Condition: Failure to isolate containment or loss of containment pressure control.

EAL:

SU8.1 Unusual Event

EITHER:

Any penetration is **not** isolated within 15 min. of a VALID containment isolation signal

OR

Containment pressure ≥ 10 psig with $<$ one full train of depressurization equipment operating (one CONTAINMENT Spray System train **AND** one CONTAINMENT Cooling System train) per design for ≥ 15 min. (Note 1)

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

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

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

RNP Basis:

First Condition:

This threshold addresses incomplete Containment isolation that allows direct release to the environment.

Second Condition:

The Containment Spray System, operating in conjunction with the Containment Cooling System, is designed to cool and depressurize the Containment structure following a Design Basis Accident (ref. 1).

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases requirement. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF

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bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, Containment Spray pump suction is transferred from the RWST to the Containment sump (ref. 2).

The Containment Cooling System consists of two trains of Containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement. Each train of two fan units is supplied with cooling water from a separate train of service water. During normal operation, all four fan units may be operating. In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically if not already running (ref. 2).

The Containment pressure setpoint (10 psig, ref. 3) is the pressure at which the equipment should actuate and begin performing its function. The design basis accident analyses and evaluations assume the loss of one ESF bus resulting in the operation of one Containment Spray System train and one Containment Cooling System train. Consistent with the design requirement, "one full train of depressurization equipment" is therefore defined to be the availability of one train of each system. If less than this equipment is operating and Containment pressure is above the actuation setpoint, the threshold is met.

NEI 99-01 Basis:

This EAL addresses a failure of one or more containment penetrations to automatically isolate (close) when required by an actuation signal. It also addresses an event that results in high containment pressure with a concurrent failure of containment pressure control systems. Absent challenges to another fission product barrier, either condition represents potential degradation of the level of safety of the plant.

For the first condition, the containment isolation signal must be generated as the result on an off-normal/accident condition (e.g., a safety injection or high containment pressure); a failure resulting from testing or maintenance does not warrant classification. The determination of containment and penetration status – isolated or not isolated – should be made in accordance with the appropriate criteria contained in the plant AOPs and EOPs. The 15-minute criterion is included to allow operators time to manually isolate the required penetrations, if possible.

The second condition addresses a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment sprays or ice condenser fans) are either lost or performing in a degraded manner.

This event would escalate to a Site Area Emergency in accordance with IC FS1 if there were a concurrent loss or potential loss of either the Fuel Clad or RCS fission product barriers.

RNP Basis Reference(s):

1. UFSAR Section 6.2.2
2. Technical Specifications Bases 3.6.6

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3. Critical Safety Function Status Tree, CSF-5 Containment
4. NEI 99-01 SU7

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Category: S – System Malfunction

Subcategory: 9 – Hazardous Event Affecting Safety Systems

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

EAL:

SA9.1 Alert

The occurrence of **any** Table S-4 hazardous event

AND EITHER:

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

| Table S-4 Hazardous Events |
|--|
| <ul style="list-style-type: none">• 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 Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

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

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

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

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

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

- (1) The integrity of the reactor coolant pressure boundary;
- (2) The capability to shut down the reactor and maintain it in a safe shutdown condition;
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

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

RNP Basis:

- The significance of seismic events are discussed under EAL HU2.1 (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2).
- The plant Seismic Category I structures are designed to withstand the effects of the design wind, 83 mph (108 gust). (ref. 3, 4).
- An explosion that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

NEI 99-01 Basis:

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

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

The second condition addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing

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SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC FS1 or RS1.

RNP Basis Reference(s):

1. AOP-021 Seismic Disturbances
2. RNP-F/PSA-0009, Assessment of Internal Flooding Events
3. UFSAR Table 3.3.1-1
4. OMM-021, Operation During Adverse Weather Conditions
5. NEI 99-01 CA6

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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 cask/canister must escape its packaging and enter the biosphere for there to be a significant environmental effect resulting from an accident involving the dry storage of spent nuclear fuel. Formal offsite planning is not required because the postulated worst-case accident involving an ISFSI has insignificant consequences to the public health and safety.

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.

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

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Category: E - ISFSI

Sub-category: None

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Notification of Unusual Event

Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask > **any** Table E-1 dose limit

| Table E-1 ISFSI Dose Limits | |
|--|---|
| 7P ISFSI | 24P ISFSI |
| <ul style="list-style-type: none">• 400 mrem/hr outside of HSM door on centerline of DSC• 400 mrem/hr at center of air inlets or outlets• 100 mrem/hr on roof, front, back or side | <ul style="list-style-type: none">• 2,600 mrem/hr on the HSM front surface• 10 mrem/hr on the HSM-H door centerline• 20 mrem/hr on the end shield wall exterior |

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 RNP ISFSI, Confinement Boundary is defined as the Dry Shielded Canister (DSC).

RNP Basis:

The ISFSIs (7P and 24P) provide for the dry storage of irradiated fuel assemblies in concrete modules. The principal components are a concrete Horizontal Storage Module (HSM) and a stainless steel Dry Shielded Canister (DSC) with an internal basket which holds the irradiated fuel assemblies. Each HSM contains one DSC and each DSC contains the spent fuel assemblies. The fuel assemblies are confined in a helium atmosphere by the stainless steel canister. Decay heat is removed by thermal radiation, conduction and convection from the canister to an air plenum inside the concrete module. Air flows through this internal plenum by natural draft convection. Both ISFSIs are totally passive system. (ref. 1).

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

The values shown in Table E-1 represent 2 times the limits specified in the respective ISFSI (7P and 24P) Technical Specification for radiation external to a loaded HSM for a DSC (ref. 1).

NEI 99-01 Basis:

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

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

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

RNP Basis Reference(s):

1. H. B. Robinson Independent Spent Fuel Storage Installation SNM-2502 License Appendix A Technical Specification Section 4.2 Limits for the Surface Dose Rate of the HSM during Storage (7P)
2. NGGM-PM-0028 contains CERTIFICATE OF COMPLIANCE NO. 1004 AMENDMENT NO. 10 (24P)
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: The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.
- B. Reactor Coolant System (RCS): The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.
- C. Containment: The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 2). “Loss” and “Potential Loss” signify the relative damage and threat of damage to the barrier. “Loss” means the barrier no longer assures containment of radioactive materials. “Potential Loss” means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Alert:

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

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

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

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- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier.
- Unusual Event ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.
- For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC RG1 has been exceeded.
- The fission product barrier thresholds specified within a scheme reflect plant-specific RNP 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 Coordinator would have more assurance that there was no immediate need to escalate to a General Emergency.

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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 Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

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

NEI 99-01 Basis:

None

RNP Basis Reference(s):

1. NEI 99-01 FA1

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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 Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

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

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

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

NEI 99-01 Basis:

None

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RNP Basis Reference(s):

1. NEI 99-01 FS1

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Category: Fission Product Barrier Degradation

Subcategory: N/A

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

EAL:

FG1.1 General Emergency

Loss of **any** two barriers

AND

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

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

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

- Loss of Fuel Clad, RCS and Containment barriers
- Loss of Fuel Clad and RCS barriers with potential loss of Containment barrier
- Loss of RCS and Containment barriers with potential loss of Fuel Clad barrier
- Loss of Fuel Clad and Containment barriers with potential loss of RCS barrier

NEI 99-01 Basis:

None

RNP Basis Reference(s):

1. NEI 99-01 FG1

ATTACHMENT 2

Fission Product Barrier Loss/Potential Loss Matrix and Bases

Introduction

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

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

- A. RCS or SG Tube Leakage
- B. Inadequate Heat removal
- C. CONTAINMENT Radiation / RCS Activity
- D. CONTAINMENT Integrity or Bypass
- E. Emergency Coordinator Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word “None” is entered in the cell.

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A would be assigned “Fuel Clad Loss A.1,” the third Containment barrier Potential Loss in Category C would be assigned “CONTAINMENT P-Loss C.3,” etc.

If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of Table F-1, locates the likely category and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine if a threshold has been exceeded. If a threshold has not been exceeded, the EAL-user proceeds to the next likely category and continues review of the thresholds in the new category

If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded,

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Fission Product Barrier Loss/Potential Loss Matrix and Bases

only that one barrier is lost or potentially lost. The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if containment radiation is sufficiently high, a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Containment barrier can occur. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, and FA1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad barrier threshold bases appear first, followed by the RCS barrier and finally the Containment barrier threshold bases. In each barrier, the bases are given according category Loss followed by category Potential Loss beginning with Category A, then B,..., E.

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

| Table F-1 Fission Product Barrier Threshold Matrix | | | | | | |
|---|---|---|---|--|---|--|
| | Fuel Clad Barrier | | Reactor Coolant System Barrier | | Containment Barrier | |
| Category | Loss | Potential Loss | Loss | Potential Loss | Loss | Potential Loss |
| A RCS or SG Tube Leakage | None | None | 1. An automatic or manual ECCS (SI) actuation required by EITHER : <ul style="list-style-type: none"> • UNISOLABLE RCS leakage • SG tube RUPTURE | 1. RCS leakage > capacity of a single charging pump (> 77 gpm) due to EITHER : <ul style="list-style-type: none"> • UNISOLABLE RCS leakage • SG tube leakage 2. CSFST Integrity- RED Path entry conditions met | 1. A leaking or RUPTURED SG is FAULTED outside of containment | None |
| B Inadequate Heat Removal | 1. CSFST Core Cooling- RED Path entry conditions met | 1. CSFST Core Cooling- ORANGE PATH entry conditions met 2. CSFST Heat Sink- RED Path entry conditions met AND Heat sink is required | None | 1. CSFST Heat Sink- RED Path entry conditions met AND Heat sink is required | None | 1. CSFST Core Cooling- RED Path entry conditions met AND Restoration procedures not effective within 15 min. (Note 1) |
| C Cont. Radiation / RCS Activity | 1. Containment High Range Radiation Monitor R-32A or R-32B > 100 R/hr 2. Dose equivalent I-131 coolant activity > 300 µCi/gm | None | 1. Containment High Range Radiation Monitor R-32A or R-32B > 5 R/hr | None | None | 1. Containment High Range Radiation Monitor R-32A or R-32B > 2000 R/hr |
| D Cont. Integrity or Bypass | None | None | None | None | 1. Containment isolation is required AND EITHER : <ul style="list-style-type: none"> • Containment integrity has been lost based on Emergency Coordinator judgment • UNISOLABLE pathway from Containment to the environment exists 2. Indications of RCS leakage outside of Containment | 1. CSFST Containment- RED Path entry conditions met 2. Containment hydrogen concentration ≥ 4% 3. Containment pressure ≥ 10 psig with < one full train of depressurization equipment operating (one CONTAINMENT Spray System train AND one CONTAINMENT Cooling System train) per design for ≥ 15 min. (Note 1) |
| E EC Judgment | 1. Any condition in the opinion of the Emergency Coordinator that indicates loss of the fuel clad barrier | 1. Any condition in the opinion of the Emergency Coordinator that indicates potential loss of the fuel clad barrier | 1. Any condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier | 1. Any condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier | 1. Any condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier | 1. Any condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier |

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: 1. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: 1. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

| |
|---|
| 1. CSFST Core Cooling-RED Path entry conditions met |
|---|

Definition(s):

None

Basis:

Plant-Specific

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path is entered if either (ref. 1):

- Core exit T/Cs are greater than or equal to 1,200°F, or
- Core exit T/Cs are greater than or equal to 700°F with RCS subcooling margin less than or equal to 18°F [37°F], no RCPs are running, and RVLIS full range is less than or equal to 41%.

CSFST values enclosed in brackets apply under adverse containment conditions, which is CV pressure greater than or equal to 4 psig (ref. 2, 3).

Either set of conditions indicates significant core exit superheating and core uncover. This is considered a Loss of the Fuel Clad barrier.

Generic

This reading indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

RNP Basis Reference(s):

1. Critical Safety Function Status Trees (CSFST), CSF-2 Core Cooling
2. OMM-022, EOP Users Guide
3. FRP-C.1, Response to Inadequate Core Cooling
4. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

| |
|--|
| 1. CSFST Core Cooling- ORANGE Path entry conditions met |
|--|

Definition(s):

None

Basis:

Plant-Specific

Critical Safety Function Status Tree (CSFST) Core Cooling-ORANGE path is entered if core exit thermocouples (T/Cs) are less than 1,200°F, RCS subcooling is less than or equal to 18°F [37°F], and any of the following (Ref. 1, 2, 3):

- No RCPs are running and either: core exit T/Cs are greater than or equal to 700°F and RVLIS full range is greater than 41%, or core exit T/Cs are less than 700°F and RVLIS full range is less than or equal to 41%.
- At least one RCP is running and Reactor Vessel water level is less than or equal to RVLIS dynamic head values specified in CSF-2 Core Cooling.

These conditions indicate subcooling has been lost and that some fuel clad damage may potentially occur.

Generic

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

RNP Basis Reference(s):

1. Critical Safety Function Status Trees, CSF-2 Core Cooling
2. FRP-C.2, Response to Degraded Core Cooling
3. OMM-022, EOP User's Guide
4. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. CSFST Heat Sink-RED Path entry conditions met

AND

Heat sink is required

Definition(s):

None

Basis:

Plant-Specific

In combination with RCS Potential Loss B.1, meeting this threshold results in a Site Area Emergency.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED path indicates the ultimate heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

Indication that heat removal is extremely challenged is manifested by entry to CSFST Heat Sink-RED path (Ref. 1, 2, 3). CSFST Heat Sink-RED path is entered if all SGs are less than or equal to 9% [18%] and total FW flow to S/Gs is less than or equal to 300 gpm or 0.2E6 lbm/hr. The combination of these conditions when heat sink is required indicates the heat sink function is under extreme challenge. This condition addresses loss of functions required for Hot Shutdown with the reactor at pressure and temperature and thus is a challenge of the Fuel Clad barrier.

The phrase “and heat sink required” precludes the need for classification for conditions in which either RCS pressure is less than SG pressure or Heat Sink-RED Path entry was created through operator action directed by an EOP. For example, FRP-H.1, Response to Loss of Secondary Heat Sink, specifically states that functional response procedure actions should not be performed if total feed flow capability of 300 gpm is available but total feed flow has been reduced due to operator action as directed by the EOPs. Therefore, Heat Sink Red Path should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification (ref. 2, 3).

Generic

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the Fuel Clad Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

RNP Basis Reference(s):

1. Critical Safety Function Status Trees, CSF-3 Heat Sink
2. FRP-H.1, Response to Loss of Secondary Heat Sink
3. OMM-022, EOP User's Guide
4. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.B

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: C. Containment Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment High Range Radiation Monitor R-32A or R-32B > 100 R/hr

Definition(s):

None

Basis:

Plant-Specific

Containment radiation monitor readings greater than 100 R/hr indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the Containment. From Calculation RNP-M/MECH-1744, "R-32A and R-32B Calculation for Core Damage Assessment," the range of calculated dose rates for 5% fuel gap release for times from 1 hour to 4 hours post reactor trip is 95 R/hr to 900 R/hr. The specified value of 100 R/hr is conservatively at the low end of the calculated range (ref. 1, 2, 3). This value is higher than that specified for RCS barrier Loss #3.

It is important to recognize that the radiation monitor may be sensitive to shine from the Reactor Vessel or RCS piping.

Monitors used for this Fission Product Barrier Loss threshold are the Containment High Range Radiation Monitors R-32A and R-32B. These monitors provide indication in the Control Room with a range of 1E0 to 1E7 R/hr (ref. 4, 5).

Generic

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300 $\mu\text{Ci/gm}$ dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

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

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

RNP Basis Reference(s):

1. EPTSC-07, Damage Assessment
2. RNP-M/MECH-1744, R-32A and R-32B Calculation for Core Damage Assessment
3. RNP-M/MECH-1745, Calculation Setpoints for Accident Rad Monitors and EP Declaration levels
4. UFSAR Section 12.3.3.1.2.2
5. OMM-014, Radiation Monitor Setpoints
5. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: C. Containment Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. Dose equivalent I-131 coolant activity > 300 $\mu\text{Ci/gm}$

Definition(s):

None

Basis:

Plant-Specific

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The threshold dose equivalent I-131 concentration is well above that expected for iodine spikes and corresponds to about 5% fuel clad damage. When reactor coolant activity reaches this level the Fuel Clad barrier is considered lost. (ref. 1).

Generic

This threshold indicates that RCS radioactivity concentration is greater than 300 $\mu\text{Ci/gm}$ 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.

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

RNP Basis Reference(s):

1. RNP-M-MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels
2. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.B

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: C. Containment Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: D. Containment Integrity or Bypass

Degradation Threat: Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: D. Containment Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: E. Emergency Coordinator Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Fuel Clad barrier

Definition(s):

None

Basis:

Plant-Specific

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

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

RNP Basis Reference(s):

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

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: E. Emergency Coordinator Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Fuel Clad barrier

Basis:

Plant-Specific

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that are to be used by the Emergency Coordinator in determining whether the Fuel Clad barrier is potentially lost. The Emergency Coordinator should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

RNP Basis Reference(s):

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

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. An automatic or manual ECCS (SI) actuation required by **EITHER:**

- UNISOLABLE RCS leakage
- SG tube RUPTURE

Definition(s):

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

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

Basis:

Plant-Specific

ECCS (SI) actuation is caused by (ref. 1):

- Pressurizer pressure < 1715 psig
- Containment pressure > 4.0 psig
- Steam Line $\Delta P > 100$ psid
- High steam flow w/ low SG pressure or low RCS T_{avg}

Generic

This threshold is based on an UNISOLABLE RCS leak of sufficient size to require an automatic or manual actuation of the Emergency Core Cooling System (ECCS). This condition clearly represents a loss of the RCS Barrier.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED. If a RUPTURED steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.

RNP Basis Reference(s):

| | | |
|----------|--------|-----------------|
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|----------|--------|-----------------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

1. EOP-E-0 Reactor Trip or Safety Injection
2. EOP-E-3 Steam Generator Tube Rupture
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Loss 1.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

1. RCS leakage > capacity of a single charging pump (> 77 gpm) due to **EITHER:**

- UNISOLABLE RCS leakage
- SG tube leakage

Definition(s):

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

Basis:

Plant-Specific

The Chemical and Volume Control System (CVCS) includes three positive displacement charging pumps each with a capacity of 77 gpm (54 - 69 gpm in the normal charging mode). RCS leakage greater than the capacity of a charging pump is indicative of substantial RCS leakage. (ref. 1, 2).

Generic

This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ECCS (SI) actuation has not occurred. The threshold is met when it is determined that RCS leakage is greater than the capacity of a single charging pump.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

If a leaking steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.

RNP Basis Reference(s):

1. AOP-16 Excessive Primary Plant Leakage
2. UFSAR Section 9.3.4 Chemical and Volume Control Center
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

2. CSFST Integrity-**RED** Path entry conditions met

Definition(s):

None

Basis:

Plant-Specific

The "Potential Loss" threshold is defined by the CSFST Reactor Coolant Integrity - RED Path. CSFST RCS Integrity - Red Path plant conditions and associated PTS Limit A indicates an extreme challenge to the safety function when plant parameters are to the left of the limit curve following excessive RCS cooldown (≥ 100 °F/hr) under pressure (ref. 1, 2).

Generic

This condition indicates an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized).

RNP Basis Reference(s):

1. CSFST CSF-4 RCS Integrity
2. CSFST CSF-4a RCS Integrity
2. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.B

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Heat Sink-RED path entry conditions met

AND

Heat sink is required

Definition(s):

None

Basis:

Plant-Specific

In combination with Fuel Clad Potential Loss B.2, meeting this threshold results in a Site Area Emergency.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED path indicates the ultimate heat sink function is under extreme challenge and that the RCS barrier may potentially be lost (ref. 1).

Indication that heat removal is extremely challenged is manifested by entry to CSFST Heat Sink-RED path (Ref. 1, 2, 3). CSFST Heat Sink-RED path is entered if all SGs are less than or equal to 9% [18%] and total FW flow to S/Gs is less than or equal to 300 gpm or 0.2E6 lbm/hr. The combination of these conditions when heat sink is required indicates the heat sink function is under extreme challenge. This condition addresses loss of functions required for Hot Shutdown with the reactor at pressure and temperature and thus is a challenge of the Fuel Clad barrier.

The phrase “and heat sink required” precludes the need for classification for conditions in which either RCS pressure is less than SG pressure or Heat Sink-RED Path entry was created through operator action directed by an EOP. For example, FRP-H.1, Response to Loss of Secondary Heat Sink, specifically states that functional response procedure actions should not be performed if total feed flow capability of 300 gpm is available but total feed flow has been reduced due to operator action as directed by the EOPs. Therefore, Heat Sink Red Path should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification (ref. 2, 3).

Generic

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the RCS Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

Meeting this threshold results in a Site Area Emergency because this threshold is identical to Fuel Clad Barrier Potential Loss threshold B.2; both will be met. This condition warrants a Site Area Emergency declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase RCS pressure to the point where mass will be lost from the system.

RNP Basis Reference(s):

1. Critical Safety Function Status Trees, CSF-3 Heat Sink
2. FRP-H.1, Response to Loss of Secondary Heat Sink
3. OMM-022, EOP User's Guide
4. NEI 99-01 Inadequate Heat Removal RCS Loss 2.B

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: C. Containment Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

| |
|---|
| 1. Containment High Range Radiation Monitor R-32A or R-32B > 5 R/hr |
|---|

Definition(s):

None

Basis:

Plant-Specific

Containment radiation monitor readings greater than 5 R/hr indicate the release of reactor coolant to the Containment. Due to the normally good fuel conditions and low RCS activity, a significant release of RCS to the Containment may result in dose rates less than the 1 R/hr minimum range of the instrument. Therefore, any positive reading on R-32A or R-32B should be considered a release of RCS to the Containment. Given that the minimum reading of the instrument is 1 R/hr and the instrument range is seven decades, 5 R/hr represents the lowest reading that is considered a clear positive response (ref. 1, 2,3).

The readings are less than those specified for Fuel Clad barrier Loss C.1 because no damage to the fuel clad is assumed. Only leakage from the RCS is assumed for this barrier Loss threshold.

It is important to recognize that the radiation monitor may be sensitive to shine from the Reactor Vessel or RCS piping. Therefore, it is possible that a reading greater than 5 Rem/hr could represent a release from fuel damage into the RCS without a release to Containment.

Monitors used for this Fission Product Barrier Loss threshold are the Containment High Range Radiation Monitors R-32A and R-32B. These monitors provide indication in the Control Room with a range of 1E0 to 1E7 R/hr (Ref. 3). Due to geometry differences, the values calculated for R-32B are approximately 80% of R-32A (ref. 1, 2).

Generic

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold C.1 since it indicates a loss of the RCS Barrier only.

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

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

RNP Basis Reference(s):

1. RNP-M/MECH-1744, R-32A and R-32B Calculation for Core Damage Assessment
2. RNP-M/MECH-1745, Calculation Setpoints for Accident Rad Monitors and EP Declaration levels
3. UFSAR Section 12.3.3.1.2.2
4. OMM-014, Radiation Monitor Setpoints
5. NEI 99-01 CMT Radiation / RCS Activity RCS Loss 3.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: B. Containment Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: D. Containment Integrity or Bypass

Degradation Threat: Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: D. Containment Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: E. Emergency Coordinator Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier

Definition(s):

None

Basis:

Plant-Specific

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the RCS barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to the recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

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

RNP Basis Reference(s):

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

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: E. Emergency Coordinator Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier

Definition(s):

None

Basis:

Plant-Specific

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the RCS barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to the inability to reach final safety acceptance criteria before completing all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that may be used by the Emergency Coordinator in determining whether the RCS Barrier is potentially lost. The Emergency Coordinator should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

RNP Basis Reference(s):

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

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. A leaking or RUPTURED SG is FAULTED outside of containment

Definition(s):

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

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

Basis:

Plant-Specific

None.

Generic

This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for RCS Barrier Potential Loss A.1 and Loss A.1, respectively. This condition represents a bypass of the containment barrier.

FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably (part of the FAULTED definition) and the FAULTED steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition, the steam generator is still considered FAULTED for emergency classification purposes.

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU4 for the fuel

ATTACHMENT 2

Fission Product Barrier Loss/Potential Loss Matrix and Bases

clad barrier (i.e., RCS activity values) and IC SU5 for the RCS barrier (i.e., RCS leak rate values).

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, gland seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

The ECLs resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

| P-to-S Leak Rate | Affected SG is FAULTED Outside of Containment? | |
|--|---|-------------------------|
| | Yes | No |
| Less than or equal to 25 gpm | No classification | No classification |
| Greater than 25 gpm | Unusual Event per SU5.1 | Unusual Event per SU5.1 |
| Requires operation of a standby charging (makeup) pump (<i>RCS Barrier Potential Loss</i>) | Site Area Emergency per FS1.1 | Alert per FA1.1 |
| Requires an automatic or manual ECCS (SI) actuation (<i>RCS Barrier Loss</i>) | Site Area Emergency per FS1.1 | Alert per FA1.1 |

There is no Potential Loss threshold associated with RCS or SG Tube Leakage.

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

RNP Basis Reference(s):

1. EOP-E-2 Faulted Steam Generator Isolation
2. EOP-E-3 Steam Generator Tube Rupture
3. NEI 99-01 RCS or SG Tube Leakage Containment Loss 1.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: B. Inadequate heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Core Cooling-RED Path entry conditions met

AND

Restoration procedures **not** effective within 15 min. (Note 1)

Definition(s):

None

Basis:

Plant-Specific

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path indicates significant core exit superheating and core uncover (ref. 1, 2).

The function restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing (ref. 2).

Generic

This condition represents an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. For this condition to occur, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.

The restoration procedure is considered “effective” if core exit thermocouple readings are decreasing and/or if reactor vessel level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The Emergency Coordinator should escalate the emergency classification level as soon as it is determined that the procedure(s) will not be effective.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation in a significant fraction of core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.

RNP Basis Reference(s):

1. CSFST CSF-2 Core Cooling
2. FRP-C.1 Response to Inadequate Core Cooling
3. NEI 99-01 Inadequate Heat Removal Containment Potential Loss 2.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: C. Containment Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: C. Containment Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

| |
|--|
| 1. Containment High Range Radiation Monitor R-32A or R-32B > 2000 R/hr |
|--|

Definition(s):

None

Basis:

Plant-Specific

Containment radiation monitor readings greater than 2000 R/hr indicate significant fuel damage, well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier. Per NEI 99-01, the desired value for Containment Potential Loss should correspond to 20% clad damage. The 2000 R/hr threshold is based on taking four times the average calculated values over the various conditions and time frames analyzed in calculation RNP-M/MECH-1744, "R-32A and R-32B Calculation for Core Damage Assessment." (ref. 1, 2).

Even though high radiation levels themselves may not represent a challenge to Containment integrity, the purpose of this criterion is to ensure precautionary public protective actions are taken due to the potential for significant public dose if the activity in the Containment were released. A reading greater than 2000 R/hr on R-31A or R-32B would result in a Loss of clad, Loss of RCS, and a Potential Loss of containment, dictating a General Emergency classification.

It is important to recognize that the radiation monitor may be sensitive to shine from the reactor vessel or RCS piping.

Monitors used for this Fission Product Barrier Loss threshold are Containment High Range Radiation Monitors R-32A and R-32B. These monitors provide indication in the Control Room with a range of 1E0 to 1E7 R/hr (Ref. 3). Due to geometry differences, the calculated values for R-32B are approximately 80% of R-32A (ref. 1, 2, 3).

Generic

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

NUREG-1228, Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the ECL to a General Emergency.

RNP Basis Reference(s):

1. RNP-M/MECH-1744, R-32A and R-32B Calculation for Core Damage Assessment
2. RNP-M/MECH-1745, Calculation Setpoints for Accident Rad Monitors and EP Declaration levels
3. UFSAR Section 12.3.3.1.2.2
4. OMM-014, Radiation Monitor Setpoints
5. NEI 99-01 CMT Radiation / RCS Activity Containment Potential Loss 3.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: D. Containment Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. Containment isolation is required

AND EITHER:

- Containment integrity has been lost based on Emergency Coordinator judgment
- UNISOLABLE pathway from containment to the environment exists

Definition(s):

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

Basis:

Plant-Specific

None

Generic

These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both bulleted thresholds.

First Threshold – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the Emergency Coordinator will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure 1. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

ATTACHMENT 2

Fission Product Barrier Loss/Potential Loss Matrix and Bases

Another example would be a loss or potential loss of the RCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

Second Threshold – Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment atmosphere to the environment. As used here, the term “environment” includes the atmosphere of a room or area, outside the containment, that may, in turn, communicate with the outside-the-plant atmosphere (e.g., through discharge of a ventilation system or atmospheric leakage). Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.

Refer to the top piping run of Figure 1. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Leakage between two interfacing liquid systems, by itself, does not meet this threshold.

Refer to the bottom piping run of Figure 1. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump developed a leak that allowed steam/water to enter the Auxiliary Building, then second threshold would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause the first threshold to be met as well.

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

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold A.1.

RNP Basis Reference(s):

1. NEI 99-01 CMT Integrity or Bypass Containment Loss 4.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: D. Containment Integrity or Bypass

Degradation Threat: Loss

Threshold:

| |
|--|
| 2. Indications of RCS leakage outside of Containment |
|--|

Definition(s):

None

Basis:

Plant-Specific

EOP-ECA-1.2 LOCA Outside Containment (ref. 1) provides instructions to identify and isolate a LOCA outside of the containment. Potential RCS leak pathways outside containment include (ref. 1):

- Residual Heat Removal
- Safety Injection
- Chemical & Volume Control
- RCP seals/seal return
- PZR/RCS sample lines

Generic

Containment sump, temperature, pressure and/or radiation levels will increase if reactor coolant mass is leaking into the containment. If these parameters have not increased, then the reactor coolant mass may be leaking outside of containment (i.e., a containment bypass sequence). Increases in sump, temperature, pressure, flow and/or radiation level readings outside of the containment may indicate that the RCS mass is being lost outside of containment.

Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of RCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside of containment may not increase significantly; however, other unexpected changes in sump levels, area temperatures or pressures, flow rates, etc. should be sufficient to determine if RCS mass is being lost outside of the containment.

Refer to the middle piping run of Figure 1. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause threshold D.1 to be met as well.

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

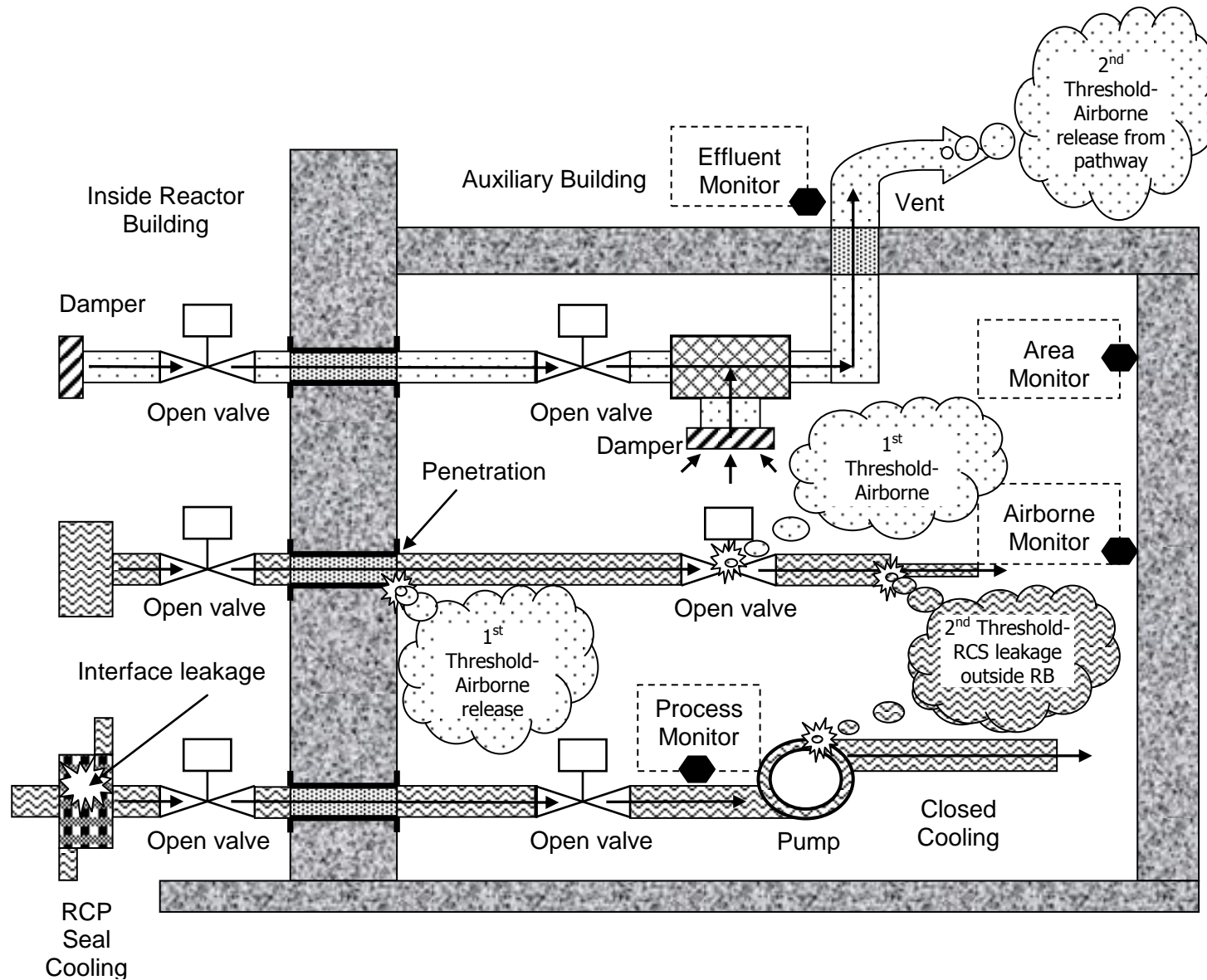
To ensure proper escalation of the emergency classification, the RCS leakage outside of containment must be related to the mass loss that is causing the RCS Loss and/or Potential Loss threshold A.1 to be met.

RNP Basis Reference(s):

1. EOP-ECA-1.2 LOCA Outside Containment
2. NEI 99-01 CMT Integrity or Bypass Containment Loss

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Figure 1: Containment Integrity or Bypass Examples



ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: D. Containment Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

1. CSFST Containment-RED Path entry conditions met

Definition(s):

None

Basis:

Plant-Specific

Critical Safety Function Status Tree (CSFST) Containment-RED Path is entered if containment pressure is greater than or equal to 42 psig and represents an extreme challenge to safety function. (ref. 1, 3).

42 psig is based on the containment design pressure (ref. 3).

Generic

If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the RCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a Site Area Emergency and General Emergency since there is now a potential to lose the third barrier.

RNP Basis Reference(s):

1. CSFST CSF-5 Containment
2. FRP-J.1 Response to High Containment Pressure
3. FSAR 6.2.1.1.5 Acceptance Criteria
4. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: D. Containment Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

| |
|---|
| 2. Containment hydrogen concentration \geq 4% |
|---|

Definition(s):

None

Basis:

Plant-Specific

If hydrogen concentration reaches the lower flammability limit of 4%, (ref. 1) in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside Containment, Loss of the Containment barrier could occur. To generate such levels of combustible gas, Loss of the Fuel Clad and RCS barriers must also have occurred. Since this threshold is also indicative of Loss of both Fuel Clad and RCS barriers with the Potential Loss of the Containment barrier, it therefore will likely warrant declaration of a General Emergency.

Two Containment hydrogen concentration monitors (with a range of 0 to 10% hydrogen) are provided on the Core Cooling and Containment Monitor in the Control Room. Hydrogen concentration is also displayed on ERFIS Points SSC-2512A and SSC-2513A (ref. 2).

Generic

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

RNP Basis Reference(s):

1. CA-3 Hydrogen Flammability in Containment
2. LP-304, Containment Hydrogen Monitor
3. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.B

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment
Category: D. Containment Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

3. Containment pressure ≥ 10 psig with $<$ one full train of depressurization equipment operating (one Containment Spray System train **AND** one Containment Cooling System train) per design for > 15 min. (Note 1)

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

Definition(s):

None

Basis:

Plant-Specific

The Containment Spray System, operating in conjunction with the Containment Cooling System, is designed to cool and depressurize the Containment structure following a Design Basis Accident (ref. 1).

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases requirement. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, Containment Spray pump suction is transferred from the RWST to the Containment sump (ref. 2).

The Containment Cooling System consists of two trains of Containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement. Each train of two fan units is supplied with cooling water from a separate train of service water. During normal operation, all four fan units may be operating. In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically if not already running (ref. 2).

The Containment pressure setpoint (10 psig, ref. 3) is the pressure at which the equipment should actuate and begin performing its function. The design basis accident analyses and evaluations assume the loss of one ESF bus resulting in the operation of one Containment Spray System train and one Containment Cooling System train. Consistent with the design requirement, "one full train of depressurization equipment" is therefore defined to be the availability of one train of each system. If less than this equipment is operating and Containment pressure is above the actuation setpoint, the threshold is met.

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Generic

This threshold describes a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems (e.g., containment sprays, ice condenser fans, etc., but not including containment venting strategies) are either lost or performing in a degraded manner.

RNP Basis Reference(s):

1. UFSAR Section 6.2.2
2. Technical Specifications Bases 3.6.6
3. Critical Safety Function Status Tree, CSF-5 Containment
4. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.C

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: F. Emergency Coordinator Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier

Definition(s):

None

Basis:

Plant-Specific

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

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

RNP Basis Reference(s):

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

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: F. Emergency Coordinator Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier

Definition(s):

None

Basis:

Plant-Specific

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

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

RNP Basis Reference(s):

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

ATTACHMENT 3

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

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.

ATTACHMENT 3

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

RNP Table R-2/H-2 Bases

NEI 99-01 Rev 06 addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant shutdown and cooldown.

The Control Room envelope is automatically placed in the emergency pressurization operating mode as a result of a Safety Injection initiation or Control Room high radiation alarm (R-1). USFSAR chapter 15 analysis demonstrates that dose to Control Room Operators remains within limits during postulated accidents. No special protection against toxic gas intrusion and no toxic gas detectors are provided in the design of the HBR 2 Control Room, however Self-contained breathing apparatuses are available in the Control Room. The buildup of toxic chemical concentrations at the Control Room air intake and within the Control Room volume was evaluated to determine the effect on Control Room habitability from postulated toxic chemical releases. UFSAR Table 6.4.4-3 summarizes the numerical results of this HBR 2 plant toxic chemical habitability analysis and shows compliance with the appropriate limits. Additionally, manual isolation capability via the Emergency Recirculation mode of operation is provided for limiting the intake of hazardous chemicals or smoke. Hazardous chemicals are not stored or transported on or near the site in sufficient quantity as to require isolation capability as a regulatory requirement, however, isolation capability is beneficial and this operational mode is included in the system design to allow the Control Room operators to isolate outside air makeup from the Control Room envelope. Based on these factors the Control Room is excluded from consideration.

Power Operation was reviewed to determine if any actions are “necessary” to maintain power operations. Over reasonable periods there are some actions outside the Control Room that are required to be performed to maintain normal operations. The following table lists the locations into which an operator may be dispatched in order perform a normal plant operation, shutdown or cooldown. The review was completed using the following procedures as the controlling documents:

GP-006-1, Normal Plant Shutdown from Power operation to Hot Shutdown, R9

GP-007, Plant Cooldown from Mode 3 to Mode 5, R101

OP-201, Residual Heat Removal System, R69

Travel paths to the locations where the equipment is operated were considered as part of the determination of affected rooms, RNP Reactor Auxiliary Building design consist of mostly single entry rooms located off of a common hallway, therefore access to the hallway is required to access a given room. Some equipment is located within the hallway itself.

ATTACHMENT 3
Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|---|---|--|--|------|
| GP-006-1 4.6.b | PLACE "B" Mixed Bed (H-OH) in service IAW OP-301-2, Chemical and Volume Control System | | REACTOR AUXILIARY BUILDING | Primary Demineralizer Room | 1 |
| GP-006-1 8.1.4 | NOTIFY E&C to shut down the RCS Zinc Injection System. | | REACTOR AUXILIARY BUILDING | Boric Acid Batch Tank room | 1 |
| GP-006-1 8.1.5 | NOTIFY E&C to control secondary chemical addition IAW CP-SEC-304, Chemical Feed System, during the power reduction. | | 1 st level Turbine Building | Secondary Sample Room & Chemical Feed Room | 1 |
| GP-006-1 8.1.11 | Startup the Auxiliary Boilers. | Associated OP requires alignment of AS system drains. | 1 st level Turbine Building, REACTOR AUXILIARY BUILDING 1 st level hallway | N/A | 1 |
| GP-006-1 8.2.1 | Adjustment of RCP Seal Injection flows (result of raising Letdown flow) | | REACTOR AUXILIARY BUILDING | Charging Pump Room | 1 |
| GP-006-1 8.2.10 | Auxiliary Boiler makeup and modulation | | 1 st level Turbine Building | N/A | 1 |
| GP-006-1 8.2.10 | TRANSFER Gland Sealing Steam to the Auxiliary Steam System IAW of OP-502 | | 1 st level & 2 nd level Turbine Building | N/A | 1 |
| GP-006-1 8.2.10 | Isolation of Extraction Steam | | 2 nd level Turbine Building | N/A | 1 |

ATTACHMENT 3

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

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|--|--|--|------|------|
| GP-006-1 8.2.11 | Cylinder heating Steam operation | | 3 rd level Turbine Building | N/A | 1 |
| GP-006-1 8.2.20.c | TRANSFER Steam Generator Blowdown to the Flash Tank with Heat Recovery Bypassed IAW OP-406. | | 1 st level & 2 nd level Turbine Building | N/A | 1 |
| GP-006-1 8.2.20.d | Verify Quenching Valve control switch is in close | | 2 nd level Turbine Building | N/A | 1 |
| GP-006-1 8.2.28.e | IF FCV-1446, CONDENSATE RECIRC, has NOT opened, THEN PERFORM the following to fail open FCV-1446 | | 1 st level Turbine Building | N/A | 1 |
| GP-006-1 8.2.30.a | Locally verify Timer Valves are open | | 3 rd level Turbine Building | N/A | 1 |
| GP-006-1 8.2.35 | Open Unit OCB disconnects | Not required to obtain shutdown or cold shutdown. Action performed in Unit 2 Switchyard, an open area outside of the Protected Area. | Unit 2 Switch Yard | N/A | 2 |
| GP-006-1 8.2.47 | Verify EH Auxiliary Filter in service / air purge | Not required to obtain shutdown or cold shutdown. | 2 nd level Turbine Building | N/A | 3 |
| GP-006-1 8.2.48 | Adjust Cylinder heating Steam | | 3 rd level Turbine Building | N/A | 3 |
| GP-006-1 8.2.49 | Isolate MS-61 / 72 if needed for excess cooldown | | 3 rd level Turbine Building | N/A | 3 |

ATTACHMENT 3

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

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|---|--|--|-------------------------------|------|
| GP-006-1 8.2.50.b | Initiate de-gas of RCS | Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Primary sample room, VCT room | 3 |
| GP-006-1 8.2.51 | Isolate SW to Generator H2 coolers | Not required to obtain shutdown or cold shutdown. Only required if Generator H2 pressure will be reduced below SW pressure and leakage exist in H2 coolers | 2 nd & 3 rd level Turbine Building | N/A | 3 |
| GP-006-1 8.2.52 | Isolate SW to secondary coolers | Not required to obtain shutdown or cold shutdown. | 1 st and 2 nd level Turbine Building | N/A | 3 |
| GP-006-1 8.2.53 | Shutdown Turbine Lube Oil | Not required to obtain shutdown or cold shutdown. | 1 st and 2 nd level Turbine Building | N/A | 3 |
| GP-006-1 8.2.54 | Caution Tag Isophase Bus Duct fans | Administrative action, Not required to obtain shutdown or cold shutdown. | 2 nd level Turbine Building | N/A | 3 |
| GP-006-1 8.2.56 | Transfer temperature control to above and below seat drains per OP-405 / adjust SGBD flow | Performed only if removing steam from secondary, not required for normal shutdown & cooldown | 1 st , 2 nd , & 3 rd level Turbine Building | N/A | 3 |
| GP-006-1 8.2.57 | If Condenser vacuum to be broken then and Polishers in service then remove from service | Performed only if removing steam from secondary, not required for normal shutdown & cooldown | 1 st level Turbine Building | Condensate polisher building | 3 |
| GP-006-1 8.2.58 | Shutdown Turbine lube oil, GS, break vacuum | Performed only if removing steam from secondary, not required for normal shutdown & cooldown | 1 st and 2 nd level Turbine Building | | 3 |

ATTACHMENT 3

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

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|---|---|--|--|-------|
| GP-006-1 8.3.16 | Isolate De-borating demineralizers | Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Primary Demineralizer Room | 3 |
| GP-006-1 8.3.20.b | Initiate OST-053 | Only initiated for Refueling Shutdown so not required for normal shutdown & cooldown | Containment – all levels | N/A | 3 |
| GP-007 5.11 | Cation or Mixed Bed demineralizer operation | | REACTOR AUXILIARY BUILDING | Primary Demineralizer Room | 3 |
| GP-007 5.13 | Initiate RCS De-gas per OP-918 | Performed prior to solid operation if RCS will be opened. Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Primary Sample Room and VCT Room | 3 |
| GP-007 6.2.2 | Prepare batches of Boric Acid | Maintain Boric Acid inventory | REACTOR AUXILIARY BUILDING | CCW Pump Room and Boric Acid Batch Tank Room | 3/4/5 |
| GP-007 6.2.8 | VERIFY that the Low Temperature Overpressure Protection System is aligned for service when greater than 350°F per OP-006. | Requires verification of pressures and operation of PORV motive force isolation valves in Containment | Containment | | 3 |
| GP-007 6.2.11.d & e | Remove Rod Drive MG sets and RPI from service | Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Rod Drive MG Set Room and 2 nd REACTOR AUXILIARY BUILDING level Hallway | 3 |
| GP-007 6.2.18.b | If SGPORV used for temperature control then station dedicated local operator | | 3 rd level Turbine Building | N/A | 3/4 |

ATTACHMENT 3

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

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|---------------------------------------|---|--|--|------|
| GP-007 6.2.22.c | Transfer GS to Auxiliary Steam | | 1 st and 2 nd level Turbine Building | N/A | 3 |
| GP-007 6.2.25.b | Close breakers for SI-865 valves | | REACTOR AUXILIARY BUILDING | 1 st level hallway and E1/E2 Room | 3 |
| GP-007 6.3.1.c/d | Install Steam Dump Jumpers | Not required to obtain shutdown or cold shutdown. Jumper allows operation of 5 dump valves, cooldown can be accomplished with 3 dump valves | E1/E2 Room | | 3 |
| GP-007 6.3.5.b | Open breakers for SI-865 valves | | REACTOR AUXILIARY BUILDING | 1 st level Hallway and E1/E2 room | 3 |
| GP-007 6.3.10.d.1 | Pulling fuses for all but one SI Pump | | REACTOR AUXILIARY BUILDING | E1/E2 Room | 3 |
| GP-007 8.3.10.d.2 | Verification of SI flowpath valves | Valves are being verified in their normal positions, Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Pipe Alley, BIT Room | 3 |
| GP-007 6.3.10.d.3 | Place Caution Tags on SI valves | Administrative action, Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Pipe Alley, BIT Room, E1/E2 Room | 3 |
| GP-007 6.4.1.f | Close breakers for SI-878A/B valves | | REACTOR AUXILIARY BUILDING | 1 st floor hallway, E1/E2 Room | 4 |
| GP-007 6.4.1.i | Place caution tags on SI valves | Administrative action, Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Pipe Alley | 4 |

ATTACHMENT 3

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

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|---|---|----------------------------------|--|------|
| GP-007 6.4.1.o/p/ q | Verify SW flow through both CCW HX / Verify both CCW HX in service / Verify 2 CCW Pumps operating | | REACTOR AUXILIARY BUILDING | CCW Pump Room | 4 |
| GP-007 8.4.1.t | Throttle CC-775, as required, to obtain desired flow through RHR HX. | | REACTOR AUXILIARY BUILDING | Spent Fuel Pump / Heat Exchanger Room | 4 |
| GP-007- 6.4.2.b | Align RHR for core cooling (OP-201) | | REACTOR AUXILIARY BUILDING | 1 st level Hallway, RHR pump room deck (access through SFP HX Room), RHR Heat Exchanger Room, Pipe Alley | 4 |
| GP-007 6.4.2.c /d | Isolate standby RHR train | | REACTOR AUXILIARY BUILDING | E1/E2 Room, 1 st level hallway | 4 |
| GP-007 6.4.3.b | Place Caution Tags on selected SI valves | Administrative action only, Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | RHR Pump Room | 4 |
| GP-007 8.4.5 | PIT testing of HCV-758 and FCV-605 | | REACTOR AUXILIARY BUILDING | RHR Heat Exchanger Room | 4 |
| GP-007 6.4.5.e | Close breakers for RHR-750 and RHR- 751 | | REACTOR AUXILIARY BUILDING | 1 st level Hallway, E1/E2 Room | 4 |
| GP-007 6.4.5.i/j/m/n/o | Observe FI-608, Cycle RHR-754A | | REACTOR AUXILIARY BUILDING | Pipe Alley | 4 |
| GP-007 6.4.5.q And 6.4.7.f | Throttle CC-748A/B | | REACTOR AUXILIARY BUILDING | RHR Heat Exchanger Room | 4 |

ATTACHMENT 3

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

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|--|--|--|--|-----------|
| GP-007 6.4.7.m | Open CVC-309E | Maximizes letdown flow, Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Non-regenerative Heat exchanger Room | 4 |
| GP-007 6.4.8.a | Locally monitor Feedwater Section Valve Cycling | Administrative action, Not required to obtain shutdown or cold shutdown. | 1 st level Turbine Building | | 4 |
| GP-007 6.4.9 | Open / Throttle MSIV above and below seat drains | | 3 rd level Turbine Building | | 4 |
| GP-007 6.4.9 | Adjust SW header pressure | | REACTOR AUXILIARY BUILDING | CCW Pump Room | 4/5 |
| GP-007 6.4.11.j | Place N2 blanket on SG per OP-406 | Not required to obtain shutdown or cold shutdown. | 3 rd level Turbine Building | | 4/5 |
| GP-007 6.4.11.k | Place clearance on SDAFW Pump to prevent MDCT overflow | Not required to obtain shutdown or cold shutdown. | 1 st level Turbine Building | N/A | 4/5 |
| GP-007 6.4.11.l | Remove Condensate polishers from service as required | Not required to obtain shutdown or cold shutdown. | 1 st level Turbine Building | Condensate polisher Building | 4/5 |
| GP-007 6.4.13.b/c | Restore RHR train maintained subcooled | | REACTOR AUXILIARY BUILDING | E1/E2 Room, 1 st level Hallway, RHR Heat Exchanger Room | 4/5 |
| GP-007 8.4.14 | Remove Rod Control and RPI from service per OP-003 | Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Rod Drive MG Set Room, 2 nd level hallway | 4/5 |
| OP-915-1 | Makeup to CST and PWST | Makeup to Condensate Storage Tank is continuous due to Steam Generator Blow Down to flash tank | 1 st level Turbine Building | Makeup water Treatment (MWT) Room and Condensate Polisher | 1,2,3,4,5 |

ATTACHMENT 3

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

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|--|--|--|--|-----------|
| | | (loss), Condensate Polishers are typically not in operation however some equipment required for MWT is located in CP building. | | (CP) Building | |
| IAO rounds | Venting of Charging Pumps | Required to prevent gas intrusion concerns | REACTOR AUXILIARY BUILDING | Charging pump Room | 1,2,3,4,5 |
| OP-106 8.2.3 | Adjustment of SGBD flow rates | | 1 st and 2 nd level Turbine Building | | 1,2,3,4 |
| OP-910 8.4.3 | Stopping / Starting of Spent Fuel Pool Cooling pumps for temperature control | Required to maintain temperature in normal control band, frequency varies depending upon lake temperature | REACTOR AUXILIARY BUILDING | Spent Fuel Pump / Heat Exchanger Room | 1,2,3,4,5 |
| OP-301-2 6.2 | Cation Bed Demineralizer Operation | Frequency varies based on RCS Boron, required to maintain RCS chemistry | REACTOR AUXILIARY BUILDING | Primary Demineralizer Room | 1,2 |
| CP-PRI-207 | RCS Sampling / Chemical Control | Maintenance of RCS chemistry, Required to verify RCS boron concentrations to support SDM for shutdown and cooldown | REACTOR AUXILIARY BUILDING | Primary Sample Room / CCW Pump Room | 1,2,3,4,5 |
| CP-SEC-304 | Secondary Sampling / Chemical Control | Required to maintain secondary chemistry | 1 st level Turbine Building | Secondary Sample Room / Chemical Feed Room | 1,2,3,4 |

ATTACHMENT 3
Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases

Table R-2/H-2 Results

| Table R-2/H-2 Safe Operation & Shutdown Rooms/Areas | |
|---|----------------|
| Room/Area | Mode(s) |
| Reactor Auxiliary Building, 1 st level hallway | 1,2,3,4,5 |
| Reactor Auxiliary Building, 2 nd level hallway | 1,2,3,4,5 |
| Charging Pump Room | 1,2,3,4,5 |
| Component Cooling Water Pump Room | 1,2,3,4,5 |
| Primary Sample Room | 1,2,3,4,5 |
| Primary Demineralizer Room | 1,2,3 |
| Spent Fuel Pump / Heat Exchanger Room | 1,2,3,4,5 |
| Pipe Alley | 4 |
| RHR Heat Exchanger Room | 4 |
| RHR Pump Room entry area (access to RHR Pump CCW flow indication / control) | 4 |
| Boric Acid Batch Tank Room | 1,2,3,4,5 |
| Emergency Bus E1/E2 Room | 3,4,5 |
| Turbine Building 1 st Floor (includes Condensate Polisher, Makeup Water Treatment and Secondary Sample Room) | 1,2,3,4 |
| Turbine Building 2 nd Floor | 1,2,3,4 |
| Turbine Building 3 rd Floor | 1,3,4 |
| Containment Building | 3 |

RNP-RA/15-0034
Enclosure 4
290 Pages (including cover page)

Enclosure 4

EMERGENCY ACTION LEVEL TECHNICAL BASES (REDLINE AND
STRIKEOUT VERSION)



Robinson Nuclear Plant

I
Information
Use

H. B. Robinson Steam Electric Plant
Plant Operating Manual
Volume 2
Part 5

EPCLA-04

***EMERGENCY ACTION LEVEL TECHNICAL
BASES DOCUMENT***

(Redline and Strikeout Version)

Revision 0 Draft 4/24/15

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1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Robinson Nuclear Plant (RNP). It should be used to facilitate review of the RNP EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EPCLA-01-110, Emergency Control, may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Coordinator in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training and for explaining event classifications to off-site officials.

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

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

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 RNP 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 (ADAMS Accession Number ML12326A805) (ref. 4.1.1), RNP conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

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

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

The primary fission product barriers are:

- A. Fuel Clad: The Fuel Clad Barrier is the zircalloy tubes that contain the fuel pellets.
- B. Reactor Coolant System (RCS): The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.
- C. Containment: The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency

2.3 Fission Product Barrier Classification Criteria

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

Alert:

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

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

2.4 EAL Organization

The RNP EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
 - EALs applicable under all plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
 - EALs applicable only under hot operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Hot Standby, Startup, or Power Operation mode.
 - EALs applicable only under cold operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling or Defueled mode.

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

- Within each 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 RNP EAL categories are aligned to and represent the NEI 99-01 "Recognition Categories." Subcategories are used in the RNP scheme as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The RNP EAL categories and subcategories are listed below.

EAL Groups, Categories and Subcategories

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

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

2.5 Technical Bases Information

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

Category Letter & Title

Subcategory Number & Title

Initiating Condition (IC)

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

EAL Identifier (enclosed in rectangle)

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

1. First character (letter): Corresponds to the EAL category as described above (R, C, H, S, F or E)
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 Operations, 2 - Startup, 3 – Hot Standby, 4 - Hot Shutdown, 5

- Cold Shutdown, 6 - Refueling, D - Defueled, or All. (See Section 2.6 for operating mode definitions)

Definitions:

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

Basis:

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

RNP Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.6 Operating Mode Applicability (ref. 4.1.7)

1 Power Operations

$K_{\text{eff}} \geq 0.99$ and reactor thermal power $> 5\%$ and average coolant temperature $\geq 350^\circ\text{F}$

2 Startup

$K_{\text{eff}} \geq 0.99$ and reactor thermal power $\leq 5\%$ average coolant temperature $\geq 350^\circ\text{F}$

3 Hot Standby

$K_{\text{eff}} < 0.99$ and average coolant temperature $\geq 350^\circ\text{F}$

3 Hot Shutdown

$K_{\text{eff}} < 0.99$ and average coolant temperature $350^\circ\text{F} > T_{\text{avg}} > 200^\circ\text{F}$ (excluding decay heat)

4 Cold Shutdown

$K_{\text{eff}} < 0.99$ and average coolant temperature $T_{\text{avg}} \leq 200^\circ\text{F}$

5 Refueling

$K_{\text{eff}} < 0.95$ and average coolant temperature $T_{\text{avg}} \leq 140^\circ\text{F}$; fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed

D Defueled

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

The plant operating mode that exists at the time that the event occurs (prior to any protective system or operator action 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 Coordinator must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes, and the informing basis information. In the Recognition Category F matrices, EALs are based on loss or potential loss of Fission Product Barrier Thresholds.

3.1.1 Classification Timeliness

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, "Interim Staff Guidance, Emergency Planning for Nuclear Power Plants" (ref. 4.1.9).

3.1.2 Valid Indications

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

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

3.1.3 Imminent Conditions

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

3.1.4 Planned vs. Unplanned Events

A planned work activity that results in an expected event or condition which meets or exceeds an EAL does not warrant an emergency declaration provided that: 1) the activity proceeds as planned, and 2) the plant remains within the limits imposed by the operating license. Such activities include planned work to test, manipulate, repair, maintain or modify a system or

component. In these cases, the controls associated with the planning, preparation and execution of the work will ensure that compliance is maintained with all aspects of the operating license provided that the activity proceeds and concludes as expected. Events or conditions of this type may be subject to the reporting requirements of 10 § CFR 50.72 (ref. 4.1.4).

3.1.5 Classification Based on Analysis

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

3.1.6 Emergency Coordinator 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 Coordinator 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 Coordinator 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. 4.1.14).

3.2.1 Classification of Multiple Events and Conditions

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

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

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

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

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

3.2.2 Consideration of Mode Changes During Classification

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

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

3.2.3 Classification of Imminent Conditions

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

3.2.4 Emergency Classification Level Upgrading and Downgrading

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

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

3.2.5 Classification of Short-Lived Events

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

3.2.6 Classification of Transient Conditions

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

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

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

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

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

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

In some cases, an EAL may be met but the emergency classification was not made at the time of the event or condition. This situation can occur when personnel discover that an event or condition existed which met an EAL, but no emergency was declared, and the event or

condition no longer exists at the time of discovery. This may be due to the event or condition not being recognized at the time or an error that was made in the emergency classification process.

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 (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 Drawing HBR2-9800, Plot Plan RNP
- 4.1.7 Technical Specifications Table 1.1-1 Modes
- 4.1.8 PRO-NGGC-0201 NGG Procedure Writers Guide
- 4.1.9 NSIR/DPR-ISG-01 Interim Staff Guidance, Emergency Planning for Nuclear Power Plants
- 4.1.10 PLP-007 Robinson Emergency Plan
- 4.1.11 UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone
- 4.1.12 OMP-003, Shutdown Safety Function Guidelines
- 4.1.13 OMM-033, Implementation of CV Closure
- 4.1.14 CM-603, Disassembly and Assembly of the Containment Equipment Hatch and Missile Barrier

4.2 Implementing

- 4.2.1 EPCLA-01, Emergency Control
- 4.2.2 NEI 99-01 Rev. 6 to RNP EAL Comparison Matrix
- 4.2.3 RNP 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 process, or have occurred, which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of hostile action. Any releases are expected to be small fractions of the EPA Protective Action Guideline exposure levels.

Confinement Boundary

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.

Containment Closure

The action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. In accordance with OMM-033, Implementation of CV Closure, Containment closure exits when (ref. 4.1.13, 4.1.14):

- The equipment hatch is closed and held in place by a sufficient number of bolts as identified in CM-603, AND
- At least one door in the personnel air lock is closed, AND
- Each penetration providing direct access from the Containment atmosphere to the outside atmosphere is either:
 - closed by a manual or automatic isolation valve, blind flange, or equivalent,
 - OR
 - capable of being closed by an OPERABLE Containment Ventilation Isolation System.

EPA PAGs

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

Explosion

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

Faulted

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

Fire

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

Flooding

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

General Emergency

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

Hostage

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

Hostile Action

An act toward RNP 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 RNP. 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.

Maintain

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

Normal Levels

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

Owner Controlled Area

That area surrounding the Protected Area beyond which RNP exercises access control.

Projectile

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

Protected Area

An area encompassed by physical barriers and to which access is controlled.
The Protected Area refers to the designated Security area around the process buildings and is depicted in Drawing HBR2-9800, Plot Plan RNP (ref. 4.1.6).

RCS Intact

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

Refueling Pathway

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

Reduced Inventory

Plant condition when fuel is in the Reactor Vessel and Reactor Coolant System level is less than or equal to -36 inches below the vessel flange (ref. 4.1.12, 4.1.13).

Ruptured

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

Restore

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

Safety System

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

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

- (1) The integrity of the reactor coolant pressure boundary;
- (2) The capability to shut down the reactor and maintain it in a safe shutdown condition;
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Security Condition

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

Site Boundary

As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone (ref. 4.1.11). For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

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.

Valid

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

Visible Damage

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

5.2 Abbreviations/Acronyms

| | |
|-----------------|--|
| °F | Degrees Fahrenheit |
| ° | Degrees |
| AC | Alternating Current |
| AOP | Abnormal Operating Procedure |
| ATWS | Anticipated Transient Without Scram |
| CDE | Committed Dose Equivalent |
| CFR | Code of Federal Regulations |
| CSFST | Critical Safety Function Status Tree |
| DBA | Design Basis Accident |
| DC | Direct Current |
| EAL | Emergency Action Level |
| EC | Emergency Coordinator |
| ECCS | Emergency Core Cooling System |
| ECL | Emergency Classification Level |
| EOF | Emergency Operations Facility |
| EOP | Emergency Operating Procedure |
| EPA | Environmental Protection Agency |
| ERG | Emergency Response Guideline |
| EPIP | Emergency Plan Implementing Procedure |
| ESF | Engineered Safety Feature |
| FAA | Federal Aviation Administration |
| FBI | Federal Bureau of Investigation |
| FEMA | Federal Emergency Management Agency |
| GE | General Emergency |
| IC | Initiating Condition |
| IPEEE | Individual Plant Examination of External Events (Generic Letter 88-20) |
| ISFSI | Independent Spent Fuel Storage Installation |
| K_{eff} | Effective Neutron Multiplication Factor |
| LCO | Limiting Condition of Operation |
| LER | Licensee Event Report |
| LOCA | Loss of Coolant Accident |
| LWR | Light Water Reactor |
| MPC | Maximum Permissible Concentration/Multi-Purpose Canister |

MSIV..... Main Steam Isolation Valve
 MSL Main Steam Line
 mR, mRem, mrem, mREM milli-Roentgen Equivalent Man
 MW Megawatt
 RCS..... Reactor Coolant System
 NEI Nuclear Energy Institute
 NESP..... National Environmental Studies Project
 NPP Nuclear Power Plant
 NRC..... Nuclear Regulatory Commission
 NSSS..... Nuclear Steam Supply System
 NORAD..... North American Aerospace Defense Command
 (NO)UE..... Notification of Unusual Event
 OBE Operating Basis Earthquake
 OCA..... Owner Controlled Area
 ODCM..... Off-site Dose Calculation Manual
 ORO Offsite Response Organization
 PA..... Protected Area
 PAG Protective Action Guideline
 PRA/PSA Probabilistic Risk Assessment / Probabilistic Safety Assessment
 PWR..... Pressurized Water Reactor
 PSIG..... Pounds per Square Inch Gauge
 R..... Roentgen
 Rem, rem, REM Roentgen Equivalent Man
 RETS..... Radiological Effluent Technical Specifications
 RNP Robinson Nuclear Plant
 RPS Reactor Protection System
 RV Reactor Vessel
 RVLIS Reactor Vessel Level Indicating System
 SAR Safety Analysis Report
 SBGTS Stand-By Gas Treatment System
 SBO..... Station Blackout
 SCBA..... Self-Contained Breathing Apparatus
 SG Steam Generator
 SI..... Safety Injection

SLC Selected Licensee Commitment
SPDS..... Safety Parameter Display System
SRO..... Senior Reactor Operator
SSF Safe Shutdown Facility
TEDE Total Effective Dose Equivalent
TOAF Top of Active Fuel
TSC Technical Support Center
UFSAR Updated Final Safety Analysis Report
WOG Westinghouse Owners Group

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

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

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

| RNP | NEI 99-01 Rev. 6 | |
|------------|-------------------------|------------------------|
| EAL | IC | Example EAL |
| 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 |
| FA1.1 | FA1 | 1 |
| FS1.1 | FS1 | 1 |
| FG1.1 | FG1 | 1 |
| HU1.1 | HU1 | 1, 2 3 |
| HU2.1 | HU2 | 1 |
| HU3.1 | HU3 | 1 |
| HU3.2 | HU3 | 2 |
| HU3.3 | HU3 | 3 |

| RNP | NEI 99-01 Rev. 6 | |
|------------|-------------------------|------------------------|
| EAL | IC | Example EAL |
| 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 |
| HG1.1 | HG1 | 1 |
| HG7.1 | HG7 | 1 |
| SU1.1 | SU1 | 1 |
| SU3.1 | SU2 | 1 |
| SU4.1 | SU3 | 2 |
| SU4.2 | SU3 | 1 |
| SU5.1 | SU4 | 1, 2, 3 |
| SU6.1 | SU5 | 1 |
| SU6.2 | SU5 | 2 |
| SU7.1 | SU6 | 1, 2, 3 |
| SU8.1 | SU7 | 1, 2 |
| SA1.1 | SA1 | 1 |

| RNP | NEI 99-01 Rev. 6 | |
|------------|-------------------------|------------------------|
| EAL | IC | Example EAL |
| SA3.1 | SA2 | 1 |
| SA6.1 | SA5 | 1 |
| SA9.1 | SA9 | 1 |
| SS1.1 | SS1 | 1 |
| SS2.1 | SS8 | 1 |
| SS6.1 | SS5 | 1 |
| SG1.1 | SG1 | 1 |
| SG1.2 | SG8 | 1 |
| EU1.1 | E-HU1 | 1 |

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Fission Product Barrier Matrix and Basis

Category R – Abnormal Rad Release / Rad Effluent

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

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

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

Events of this category pertain to the following subcategories:

1. Radiological Effluent

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

2. Irradiated Fuel Event

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

3. Area Radiation Levels

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

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.1 Unusual Event

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

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

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

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

| Table R-1 Effluent Monitor Classification Thresholds | | | | | | |
|--|-----------------------|------------|-------------|---------------|---------------|-------------|
| Release Point | | Monitor | GE | SAE | Alert | UE |
| Gaseous | Plant Vent | R-14C | --- | --- | --- | 2.16E+5 cpm |
| | | R-14D | 6.38E+5 cpm | 6.38E+4 cpm | 6.38E+3 cpm | --- |
| | | R-14E | 3.31E+3 cpm | 3.40E+2 cpm | 4.30E+1 cpm | --- |
| | FHB Exhaust | R-20 | --- | --- | --- | 8.06E+5 cpm |
| | FHB Exhaust HR | R-30 | --- | 2.60E+4 mR/hr | 2.60E+3 mR/hr | --- |
| Liquid | Liquid Waste Disposal | R-18 | ---- | ---- | ---- | 4.08E+6 cpm |
| | SGBD Effluent | R-19 A/B/C | ---- | ---- | ---- | 6.94E+5 cpm |
| | Condensate Polisher | R-37 | ---- | ---- | ---- | 4.23E+5 cpm |

Mode Applicability:

All

Definition(s):

None

RNP Basis:

The column "UE" gaseous and liquid release values in Table R-1 represents two times the alarm setpoint of the specified monitors. The setpoints are established to ensure the ODCM release limits are not exceeded (ref. 1, 2).

NEI 99-01 Basis:

| | | |
|----------|--------|----------------|
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|----------|--------|----------------|

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

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

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

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

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

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

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

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

Escalation of the emergency classification level would be via IC AA1RA1.

RNP Basis Reference(s):

1. RNP-M/MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels for H. B. Robinson, Unit No. 2
2. EP-EALCALC-RNP-1401, RNP Radiological Effluent EAL Values
3. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.2 Unusual Event

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

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

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

Mode Applicability:

All

Definition(s):

None

RNP Basis:

Releases in excess of two times the site Offsite Dose Calculation Manual (ODCM) (ref. 1) instantaneous limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

NEI 99-01 Basis:

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

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

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

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

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

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

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

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

Escalation of the emergency classification level would be via IC AA1RA1.

RNP Basis Reference(s):

1. H. B. Robinson Steam Electric Plant, Unit No. 2, Off-Site Dose Calculation Manual
2. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.1 Alert

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

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

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

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

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

| Table R-1 Effluent Monitor Classification Thresholds | | | | | | |
|--|-----------------------|------------|-------------|---------------|---------------|-------------|
| Release Point | | Monitor | GE | SAE | Alert | UE |
| Gaseous | Plant Vent | R-14C | --- | --- | --- | 2.16E+5 cpm |
| | | R-14D | 6.38E+5 cpm | 6.38E+4 cpm | 6.38E+3 cpm | --- |
| | | R-14E | 3.31E+3 cpm | 3.40E+2 cpm | 4.30E+1 cpm | --- |
| | FHB Exhaust | R-20 | --- | --- | --- | 8.06E+5 cpm |
| | FHB Exhaust HR | R-30 | --- | 2.60E+4 mR/hr | 2.60E+3 mR/hr | --- |
| Liquid | Liquid Waste Disposal | R-18 | ---- | ---- | ---- | 4.08E+6 cpm |
| | SGBD Effluent | R-19 A/B/C | ---- | ---- | ---- | 6.94E+5 cpm |
| | Condensate Polisher | R-37 | ---- | ---- | ---- | 4.23E+5 cpm |

Mode Applicability:

All

Definition(s):

None

RNP Basis:

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

- 10 mRem TEDE
- 50 mRem CDE Thyroid

The column "ALERT" gaseous effluent release values in Table R-1 correspond to calculated doses of 1% (10% of the calculated SAE thresholds) of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1, 2).

NEI 99-01 Basis:

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

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

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

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

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

RNP Basis Reference(s):

1. RNP-M/MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels for H. B. Robinson, Unit No. 2
2. EP-EALCALC-RNP-1401, RNP Radiological Effluent EAL Values
3. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.2 Alert

Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 3, 4)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

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

NEI 99-01 Basis:

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

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

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

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

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

RNP Basis Reference(s):

1. AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.3 Alert

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

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

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

NEI 99-01 Basis:

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

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

addresses the spectrum of possible accident events and conditions.

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

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

Escalation of the emergency classification level would be via IC ~~AS1RS1~~.

RNP Basis Reference(s):

1. H. B. Robinson Steam Electric Plant, Unit No. 2, Off-Site Dose Calculation Manual
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.4 Alert

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

EPRAD-01, Environmental Monitoring provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

NEI 99-01 Basis:

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

release).

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

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

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

Escalation of the emergency classification level would be via IC ~~AS1RS1~~.

RNP Basis Reference(s):

1. EPRAD-01, Environmental Monitoring
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.1 Site Area Emergency

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

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

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

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

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

| Table R-1 Effluent Monitor Classification Thresholds | | | | | | |
|--|-----------------------|------------|-------------|---------------|---------------|-------------|
| Release Point | | Monitor | GE | SAE | Alert | UE |
| Gaseous | Plant Vent | R-14C | --- | --- | --- | 2.16E+5 cpm |
| | | R-14D | 6.38E+5 cpm | 6.38E+4 cpm | 6.38E+3 cpm | --- |
| | | R-14E | 3.31E+3 cpm | 3.40E+2 cpm | 4.30E+1 cpm | --- |
| | FHB Exhaust | R-20 | --- | --- | --- | 8.06E+5 cpm |
| | FHB Exhaust HR | R-30 | --- | 2.60E+4 mR/hr | 2.60E+3 mR/hr | --- |
| Liquid | Liquid Waste Disposal | R-18 | ---- | ---- | ---- | 4.08E+6 cpm |
| | SGBD Effluent | R-19 A/B/C | ---- | ---- | ---- | 6.94E+5 cpm |
| | Condensate Polisher | R-37 | ---- | ---- | ---- | 4.23E+5 cpm |

Mode Applicability:

All

Definition(s):

None

RNP Basis:

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

- 100 mRem TEDE
- 500 mRem CDE Thyroid

The column “SAE” gaseous effluent release value in Table R-1 corresponds to calculated doses of 10% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1, 2).

NEI 99-01 Basis:

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

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

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

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

Escalation of the emergency classification level would be via IC **AG4RG1**.

RNP Basis Reference(s):

1. RNP-M/MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels for H. B. Robinson, Unit No. 2
2. EP-EALCALC-RNP-1401, RNP Radiological Effluent EAL Values
3. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.2 Site Area Emergency

Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 3, 4)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

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

NEI 99-01 Basis:

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

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

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid

CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

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

Escalation of the emergency classification level would be via IC ~~AG4~~RG1.

RNP Basis Reference(s):

1. AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.3 Site Area Emergency

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

EPRAD-01, Environmental Monitoring provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

NEI 99-01Basis:

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 AG1RG1.

RNP Basis Reference(s):

1. EPRAD-01, Environmental Monitoring
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.1 General Emergency

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

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

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

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

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

| Table R-1 Effluent Monitor Classification Thresholds | | | | | | |
|--|-----------------------|------------|-------------|---------------|---------------|-------------|
| Release Point | | Monitor | GE | SAE | Alert | UE |
| Gaseous | Plant Vent | R-14C | --- | --- | --- | 2.16E+5 cpm |
| | | R-14D | 6.38E+5 cpm | 6.38E+4 cpm | 6.38E+3 cpm | --- |
| | | R-14E | 3.31E+3 cpm | 3.40E+2 cpm | 4.30E+1 cpm | --- |
| | FHB Exhaust | R-20 | --- | --- | --- | 8.06E+5 cpm |
| | FHB Exhaust HR | R-30 | --- | 2.60E+4 mR/hr | 2.60E+3 mR/hr | --- |
| Liquid | Liquid Waste Disposal | R-18 | ---- | ---- | ---- | 4.08E+6 cpm |
| | SGBD Effluent | R-19 A/B/C | ---- | ---- | ---- | 6.94E+5 cpm |
| | Condensate Polisher | R-37 | ---- | ---- | ---- | 4.23E+5 cpm |

Mode Applicability:

All

Definition(s):

None

RNP Basis:

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

- 1000 mRem TEDE
- 5000 mRem CDE Thyroid

The column “GE” gaseous effluent release values in Table R-1 correspond to calculated doses of 100% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1, 2).

NEI 99-01Basis:

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.

RNP Basis Reference(s):

1. RNP-M/MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels for H. B. Robinson, Unit No. 2
2. EP-EALCALC-RNP-1401, RNP Radiological Effluent EAL Values
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.2 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 (Notes 3, 4)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

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

NEI 99-01 Basis:

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

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

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

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

RNP Basis Reference(s):

1. AD-EP-ALL-0202, Emergency Response Offsite Dose Assessment
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.3 General Emergency

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

EPRAD-01, Environmental Monitoring provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

NEI 99-01 Basis:

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

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

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

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

RNP Basis Reference(s):

1. EPRAD-01, Environmental Monitoring
2. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Unplanned loss of water level above irradiated fuel

EAL:

RU2.1 Unusual Event

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

AND

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

- R-2 CV Area
- R-5 Spent Fuel Pit Area
- Local area survey

Mode Applicability:

All

Definition(s):

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

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

RNP Basis:

The low water level alarm in this EAL refers to the Spent Fuel Pit (SFP) low level alarm. The fuel transfer canal is normally in communication with the spent fuel pit. During refueling operations, the refueling cavity in the Containment is filled and is in communication with the fuel transfer canal when the fuel transfer tube gate valve is open. A decrease in water level in the SFP, fuel transfer canal or refueling cavity is therefore sensed by the SFP low level alarm. Neither the refueling cavity, nor the fuel transfer canal, is equipped with a low level alarm.

The specified radiation monitors are those expected to see increase area radiation levels as a result of a loss of REFUELING PATHWAY inventory (ref. 4, 5, 6). Increasing radiation indications on these monitors in the absence of indications of decreasing REFUELING CAVITY level are not classifiable under this EAL.

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

The SFP level is remotely monitored by level indicator LA-651. The level switch initiates high and low level annunciators. The Spent Fuel Pit Low Level alarm (APP-036-B6) actuates if SFP level decreases to the 36 ft. 2.5 in. (Ref. 5). In addition, the Radiation Control personnel have cameras for Containment and fuel handling building remote monitoring providing visual indication of low pool or cavity levels.

Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment. Technical Specifications Section 3.7.12 (Ref. 2) requires at least 21 ft. of water above irradiated fuel in the spent fuel pit storage racks. Technical Specifications LCO 3.9.6 (Ref. 3) requires at least 23 ft. of water above the reactor vessel flange. During refueling, this maintains sufficient water level in the fuel transfer canal, refueling cavity, and SFP to retain iodine fission product activity in the water in the event of a fuel handling accident.

While radiation monitors (e.g., R-5 Spent Fuel Pit Area Radiation Monitor or portable survey instrument) could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication, in and of itself, of whether or not inventory is being lost. Generally, elevated radiation monitor indications need to be combined with another indicator (or personnel report) of water loss.

NEI 99-01 Basis:

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

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

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

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

Escalation of the emergency classification level would be via IC **AA2RA2**.

RNP Basis Reference(s):

1. USAR Section 9.1.2, Spent Fuel Storage
2. TS Section 3.7.12
3. LCO 3.9.6, Refueling Cavity Water Level
4. AOP-036, SFP Events
5. APP-036-B6, Spent Fuel Pit Low Level
6. AOP-005, Radiation Monitoring System
7. NEI 99-01 AU2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.1 Unusual Event

Uncovery of irradiated fuel in the REFUELING PATHWAY

Mode Applicability:

All

Definition(s):

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

RNP Basis:

None.

NEI 99-01 Basis:

—This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool ~~(see Developer Notes)~~. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant. ~~This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC E-HU1.~~

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

EAL #1

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

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

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

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

———EAL #3

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

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

RNP Basis Reference(s):

1. AOP-013 Fuel Handling Accident
2. AOP-036 SFP Events
3. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity

AND

A high alarm on **any** of the following:

- R-2 CV Area
- R-5 Spent Fuel Pit Area
- R-11/R-12 Process Monitor CV Air and Plant Vent (when sampling CV)
- R-14 Plant Vent
- R-21 Fuel Handling Building Upper Level

Mode Applicability:

All

Definition(s):

None

RNP Basis:

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

The high alarm setpoints for the radiation monitors are set to be indicative of significant increases in area and/or airborne radiation (ref. 4).

NEI 99-01 Basis:

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

~~————This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the~~

~~CONFINEMENT BOUNDARY is classified in accordance with IC E-HU1.~~

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

~~——— EAL # This EAL escalates from AU2 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncover of irradiated fuel. Indications of irradiated fuel uncover 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 assemblies stored in the pool.~~

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

RNP Basis Reference(s):

1. AOP-013 Fuel Handling Accident
2. AOP-036 SFP Events
3. AOP-005 Radiation Monitoring System
4. OMM-014, Radiation Monitor Setpoints
5. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.3 Alert

Lowering of spent fuel pool level to ≤ 24 ft.

Mode Applicability:

All

Definition(s):

None

RNP Basis:

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

The SFP level instruments consist of a primary channel (LI-11442A & LI-11443A) and back-up channel (LI-11442B & LI-11443B) each spanning approximately 24 ft. (14 ft. – 38 ft. indicated). Level 2 corresponds to an indicated SFP level of 24 ft. or approximately 10 ft. above the top of the SFP racks (ref. 2).

NEI 99-01 Basis:

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

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

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

~~———— EAL # This EAL escalates from AU2 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncover of irradiated fuel. Indications of irradiated fuel uncover 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 assemblies stored in the pool.

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

RNP Basis Reference(s):

1. NRC EA-12-051 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. Engineering Change EC89580
3. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RS2.1 Site Area Emergency

Lowering of spent fuel pool level to ≤ 14.75 ft.

Mode Applicability:

All

Definition(s):

None

RNP Basis:

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

The SFP level instruments consist of a primary channel (LI-11442A & LI-11443A) and back-up channel (LI-11442B & LI-11443B) each spanning approximately 24 ft. (14 ft. – 38 ft. indicated). Level 3 (top of the spent fuel racks) corresponds to an SFP level of 14 ft. However, the level instruments can actually only measure to 14.75 ft. (ref. 2).

NEI 99-01 Basis:

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

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

Escalation of the emergency classification level would be via IC ~~AG1-RG1~~ or ~~AG2RG2~~.

RNP Basis Reference(s):

1. NRC EA-12-051 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. Engineering Change EC89580
3. NEI 99-01 AS2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RG2.1 General Emergency

Spent fuel pool level cannot be restored to at least 14.75 ft. for ≥ 60 min.
(Note 1)

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

Mode Applicability:

All

Definition(s):

None

RNP Basis:

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

The SFP level instruments consist of a primary channel (LI-11442A & LI-11443A) and back-up channel (LI-11442B & LI-11443B) each spanning approximately 24 ft. (14 ft. – 38 ft. indicated). Level 3 (top of the spent fuel racks) corresponds to an SFP level of 14 ft. However, the level instruments can actually only measure to 14.75 ft (ref. 2).

NEI 99-01 Basis:

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

RNP Basis Reference(s):

1. NRC EA-12-051 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation

- 2. Engineering Change EC89580
- 3. NEI 99-01 AG2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

RA3.1 Alert

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

Control Room (R-1)

OR

Central Alarm Station (by survey)

Mode Applicability:

All

Definition(s):

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

RNP Basis:

Areas that meet this threshold include the Control Room and the Central Alarm Station (CAS). Area radiation monitor (ARM) R-1 monitors the Control room for area radiation (ref. 1, 2). The CAS is included in this EAL because of its' importance to permitting access to areas required to assure safe plant operations.

There is no permanently installed CAS area radiation monitors that may be used to assess this EAL threshold. Therefore this threshold must be assessed via local radiation survey for the CAS.

NEI 99-01 Basis:

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The **Emergency Director** Emergency Coordinator should consider the cause of the increased radiation levels and determine if another IC may be applicable. ~~For EAL #2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant~~

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

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

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

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

RNP Basis Reference(s):

1. OMM-014, Radiation Monitor Setpoints
2. DBD-SD-19 Radiation Monitoring System
3. NEI 99-01 AA3

ATTACHMENT 1 EAL Bases

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

RA3.2 Alert

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

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

| Table R-2/H-2 Safe Operation & Shutdown Rooms/Areas | |
|---|----------------|
| Room/Area | Mode(s) |
| Reactor Auxiliary Building, 1 st level hallway | 1,2,3,4,5 |
| Reactor Auxiliary Building, 2 nd level hallway | 1,2,3,4,5 |
| Charging Pump Room | 1,2,3,4,5 |
| Component Cooling Water Pump Room | 1,2,3,4,5 |
| Primary Sample Room | 1,2,3,4,5 |
| Primary Demineralizer Room | 1,2,3 |
| Spent Fuel Pump / Heat Exchanger Room | 1,2,3,4,5 |
| Pipe Alley | 4 |
| RHR Heat Exchanger Room | 4 |
| RHR Pump Room entry area (access to RHR Pump CCW flow indication / control) | 4 |
| Boric Acid Batch Tank Room | 1,2,3,4,5 |
| Emergency Bus E1/E2 Room | 3,4,5 |
| Turbine Building 1 st Floor (includes Condensate Polisher, Makeup Water Treatment and Secondary Sample Room) | 1,2,3,4 |
| Turbine Building 2 nd Floor | 1,2,3,4 |
| Turbine Building 3 rd Floor | 1,3,4 |
| Containment Building | 3 |

Mode Applicability:

All

Definition(s):

ATTACHMENT 1

EAL Bases

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.

RNP Basis:

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

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

NEI 99-01 Basis:

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

For **EAL-#2RA3.2**, 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.

ATTACHMENT 1

EAL Bases

- The increased radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

Escalation of the emergency classification level would be via Recognition Category **AR**, C or F ICs.

RNP Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases
2. NEI 99-01 AA3

ATTACHMENT 1
EAL Bases

Category C – Cold Shutdown / Refueling System Malfunction

EAL Group: Cold Conditions (RCS temperature $\leq 200^{\circ}\text{F}$); EALs in this category are applicable only in one or more cold operating modes.

Category C EALs are directly associated with cold shutdown or refueling system safety functions. Given the variability of plant configurations (e.g., systems out-of-service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The cold shutdown and refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, containment closure, and fuel clad integrity for the applicable operating modes (5 - Cold Shutdown, 6 - Refueling, D – Defueled).

The events of this category pertain to the following subcategories:

1. RCS Level

Reactor Vessel or RCS water level is directly related to the status of adequate core cooling and, therefore, fuel clad integrity.

2. Loss of Emergency AC Power

Loss of Emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite power sources for 480V emergency buses.

3. RCS Temperature

Uncontrolled or inadvertent temperature or pressure increases are indicative of a potential loss of safety functions.

4. Loss of Vital DC Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of power to or degraded voltage on the 125 VDC safeguard buses.

ATTACHMENT 1
EAL Bases

5. Loss of Communications

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

6. Hazardous Event Affecting Safety Systems

Certain hazardous natural and technological events may result in visible damage to or degraded performance of safety systems warranting classification.

ATTACHMENT 1

EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory for 15 minutes or longer

EAL:

CU1.1 Unusual Event

UNPLANNED loss of reactor coolant results in RCS water level less than a required lower limit for ≥ 15 min. (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

RNP Basis:

RCS water level less than a required lower limit is meant to be less than the lower end of the level control band being procedurally maintained for the current condition or evolution.

With the plant in Cold Shutdown, RCS water level is normally maintained above the pressurizer low level setpoint (ref. 1, 3). However, if RCS level is being controlled below the pressurizer low level setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

With the plant in Refueling mode, RCS water level is normally maintained at or above the reactor vessel flange (ref. 2, 3, 4, 5).

NEI 99-01 Basis:

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor ~~(reactor vessel/RCS [PWR] or RPV [BWR])~~ level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required

ATTACHMENT 1

EAL Bases

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.

RNP Basis Reference(s):

1. APP- 003 RCS & Makeup Systems
2. GP-001 Fill and Vent of the Reactor Coolant System
3. GP-008 Draining the Reactor Coolant System
4. GP-009-3 Draining the Refueling Cavity With Fuel in the Reactor Vessel
5. GP-009-5 Adjusting Reactor Vessel Level After Refueling Cavity Drain With Fuel In the Reactor
6. NEI 99-01 CU1

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory for 15 minutes or longer

EAL:

CU1.2 Unusual Event

RCS water level cannot be monitored

AND EITHER

- UNPLANNED increase in **any** Table C-1 sump or tank due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

| Table C-1 Sumps / Tanks |
|---|
| <ul style="list-style-type: none">• Containment (CV) sump• PRT• RCDT• CCW surge tank |

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

ATTACHMENT 1 EAL Bases

RNP Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available. RCS level in the Refueling mode is normally monitored using the standpipe.

In this EAL, all water level indication is unavailable and the RCS inventory loss must be detected by indirect leakage indications. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Sumps and tanks where RCS leakage may accumulate are listed in Table C-1. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2, 3, 4).

NEI 99-01 Basis:

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor ~~(reactor vessel/RCS [PWR] or RPV [BWR])~~ level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

~~———EAL #1 recognizes that the minimum required (reactor vessel/RCS [PWR] or RPV [BWR]) level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.~~

~~The 15-minute threshold duration allows sufficient time for prompt operator actions to restore and maintain the expected water level. This criterion excludes transient conditions causing a brief lowering of water level.~~

This EAL #2 addresses a condition where all means to determine ~~(reactor vessel/RCS [PWR] or RPV [BWR])~~ level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the ~~(reactor vessel/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.

RNP Basis Reference(s):

| | | |
|----------|--------|----------------|
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|----------|--------|----------------|

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EAL Bases

1. GP-001 Fill and Vent of the Reactor Coolant System
2. GP-008 Draining the Reactor Coolant System
3. GP-009-3 Draining the Refueling Cavity With Fuel in the Reactor Vessel
4. GP-009-5 Adjusting Reactor Vessel Level After Refueling Cavity Drain With Fuel In the Reactor
5. NEI 99-01 CU1

ATTACHMENT 1

EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory

EAL:

CA1.1 Alert

Loss of RCS inventory as indicated by RCS water level < -72 in. (69% RVLIS Full Range)

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

None

RNP Basis:

When reactor vessel water level decreases to < -72 in. (< 69% RVLIS Full Range) (ref. 1, 2), RHR pumps must be tripped. If level is below -72 inches, vortexing and air entrainment may result in damage to the RHR Pumps.

NEI 99-01 Basis:

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

For this EAL #1, a lowering of RCS water level below ~~-72 inches(site-specific level)-ft.~~ indicates that operator actions have not been successful in restoring and maintaining ~~(reactor vessel/RCS [PWR] or RPV [BWR])~~ water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncover.

Although related, this EAL #1 is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a ~~Residual Decay~~ Heat Removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

~~For 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~~

ATTACHMENT 1
EAL Bases

~~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 ~~water~~ level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

RNP Basis Reference(s):

1. GP-008, Draining the Reactor Coolant System
2. AOP-020 Loss of Residual Heat Removal (Shutdown Cooling)
3. NEI 99-01 CA1

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory

EAL:

CA1.2 Alert

RCS water level cannot be monitored for ≥ 15 min. (Note 1)

AND EITHER

- UNPLANNED increase in **any** Table C-1 sump or tank due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

| Table C-1 Sumps / Tanks |
|---|
| <ul style="list-style-type: none">• Containment (CV) sump• PRT• RCDT• CCW surge tank |

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

RNP Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. In the Refuel mode, the RCS is not intact and RPV level may be monitored by different means, including the ability to monitor level visually.

ATTACHMENT 1

EAL Bases

In this EAL, all RCS water level indication would be unavailable for greater than 15 minutes, and the RCS inventory loss must be detected by indirect leakage indications. Sump level increases must be evaluated against other potential sources of leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Sumps and tanks where RCS leakage may accumulate are listed in Table C-1. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2, 3, 4).

NEI 99-01 Basis:

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

~~For EAL #1, a lowering of water level below (site-specific level) indicates that operator actions have not been successful in restoring and maintaining (reactor vessel/RCS [PWR] or RPV [BWR]) water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncover.~~

~~Although related, EAL #1 is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Residual Heat Removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.~~

For this EAL #2, the inability to monitor ~~(reactor vessel/RCS [PWR] or RPV [BWR])~~ level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the ~~(reactor vessel/RCS [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.

RNP Basis Reference(s):

1. GP-001 Fill and Vent of the Reactor Coolant System
2. GP-008 Draining the Reactor Coolant System
3. GP-009-3 Draining the Refueling Cavity With Fuel in the Reactor Vessel
4. GP-009-5 Adjusting Reactor Vessel Level After Refueling Cavity Drain With Fuel In the Reactor
5. NEI 99-01 CA1

ATTACHMENT 1
EAL Bases

ATTACHMENT 1

EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory affecting core decay heat removal capability

EAL:

| |
|----------------------------------|
| CS1.1 Site Area Emergency |
|----------------------------------|

| |
|---|
| With CONTAINMENT CLOSURE not established, RCS level < 64.5% RVLIS Full Range |
|---|

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. In accordance with OMM-033, Implementation of CV Closure, Containment closure exits when:

- The equipment hatch is closed and held in place by a sufficient number of bolts as identified in CM-603, AND
- At least one door in the personnel air lock is closed, AND
- Each penetration providing direct access from the Containment atmosphere to the outside atmosphere is either:
 - closed by a manual or automatic isolation valve, blind flange, or equivalent,
OR
 - capable of being closed by an OPERABLE Containment Ventilation Isolation System.

RNP Basis:

64.5% RVLIS Full Range corresponds to the level of six inches below the bottom ID of the RCS hot leg penetration (240' 7" el.) (ref. 1, 2).

Six inches below the elevation of the bottom of the RCS hot leg penetration can be monitored only by RVLIS full range (64.5%). Level monitoring instruments LI-403, LI-404, Standpipe Loop B, and Standpipe Loop C cannot sense level changes in the reactor vessel below the elevation of the RCS loop hot leg penetration. The RVLIS full range threshold has been determined as follows (ref. 2, 3):

ATTACHMENT 1 EAL Bases

| Component Dimensions | | RVLIS Full Range (%) |
|---|----------|----------------------|
| Reactor Vessel bottom head OD to top of Control Rod Mechanism housing (in.) | 498.000 | NA |
| Thickness of bottom head (in.) | 5.187 | NA |
| Thickness of vessel head (in.) | 7.750 | NA |
| Height of Control Rod Mechanism above vessel closure head (in.) | 18.000 | NA |
| Inner height of vessel (in.): $498.000 - 5.187 - 7.750 - 18.000 =$ | 467.063 | 100.0 |
| Bottom of vessel (in.) | 0.000 | 0.0 |
| RVLIS span %/in.: $(100.0 - 0.0)/(467.063 - 0.000) =$ | 0.214 | NA |
| Height of RCS hot leg centerline above vessel bottom (in.) | 321.7813 | NA |
| RCS hot leg penetration diameter (in.) | 29.000 | NA |
| Height of bottom of RCS hot leg above vessel bottom (in.): $321.783 - (29.000/2) =$ | 307.2813 | A |
| 6 in. below height of bottom of hot leg (in.): $307.2813 - 6 =$ | 301.2813 | B |
| Height of top of fuel above vessel bottom (in.) | 279.5313 | C |

A = Height of bottom of RCS hot leg above vessel bottom x RVLIS span = 65.8%

B = 6 in. below height of bottom of hot leg x RVLIS span = 64.5%

C = Height of top of fuel above vessel bottom x RVLIS span = 59.8%

At RNP RVLIS is normally not available when in Mode 6. The RVLIS connection to the reactor vessel head is removed prior to removing the reactor vessel head. Under those conditions where RCS level cannot be monitored, classification should be made based on CS1.3.

NEI 99-01 Basis:

This IC addresses a significant and prolonged loss of ~~(reactor vessel/RCS- [PWR] or RPV [BWR])~~ inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RCS/~~reactor vessel~~ level cannot be restored, fuel damage is probable.

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS/~~reactor vessel~~ levels of ~~EALs 1.b~~CS1.1 and ~~2.b~~CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

~~In EAL 3.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncover has actually occurred (i.e., to~~

ATTACHMENT 1

EAL Bases

~~account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.~~

~~The inability to monitor (reactor vessel/RCS [PWR] or RPV [BWR]) level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the (reactor vessel/RCS [PWR] or RPV [BWR]).~~

~~These~~ This EALs 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 AG4RG1

RNP Basis Reference(s):

1. GP-008, Draining the Reactor Coolant System
2. UFSAR Table 5.3.0-1
3. UFSAR Figure 5.3.0-1
4. NEI 99-01 CS1

ATTACHMENT 1

EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory affecting core decay heat removal capability

EAL:

| |
|----------------------------------|
| CS1.2 Site Area Emergency |
|----------------------------------|

| |
|--|
| With CONTAINMENT CLOSURE established, RCS level < 59.8% RVLIS Full Range |
|--|

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. In accordance with OMM-033, Implementation of CV Closure, Containment closure exits when:

- The equipment hatch is closed and held in place by a sufficient number of bolts as identified in CM-603, AND
- At least one door in the personnel air lock is closed, AND
- Each penetration providing direct access from the Containment atmosphere to the outside atmosphere is either:
 - closed by a manual or automatic isolation valve, blind flange, or equivalent,
 - OR
 - capable of being closed by an OPERABLE Containment Ventilation Isolation System.

RNP Basis:

59.8% RVLIS Full Range corresponds to the top of active fuel. Other RCS level instruments are off-scale low when core uncover occurs (ref. 1, 2).

When reactor vessel water level drops below the RVLIS full range setpoint of 59.8%, core uncover is about to occur. The RVLIS full range threshold has been determined as follows (ref. 2, 3):

ATTACHMENT 1

EAL Bases

| Component Dimensions | | RVLIS Full Range (%) |
|---|----------|----------------------|
| Reactor Vessel bottom head OD to top of Control Rod Mechanism housing (in.) | 498.000 | NA |
| Thickness of bottom head (in.) | 5.187 | NA |
| Thickness of vessel head (in.) | 7.750 | NA |
| Height of Control Rod Mechanism above vessel closure head (in.) | 18.000 | NA |
| Inner height of vessel (in.): $498.000 - 5.187 - 7.750 - 18.000 =$ | 467.063 | 100.0 |
| Bottom of vessel (in.) | 0.000 | 0.0 |
| RVLIS span %/in.: $(100.0 - 0.0)/(467.063 - 0.000) =$ | 0.214 | NA |
| Height of RCS hot leg centerline above vessel bottom (in.) | 321.7813 | NA |
| RCS hot leg penetration diameter (in.) | 29.000 | NA |
| Height of bottom of RCS hot leg above vessel bottom (in.): $321.783 - (29.000/2) =$ | 307.2813 | A |
| 6 in. below height of bottom of hot leg (in.): $307.2813 - 6 =$ | 301.2813 | B |
| Height of top of fuel above vessel bottom (in.) | 279.5313 | C |

A = Height of bottom of RCS hot leg above vessel bottom x RVLIS span = 65.8%

B = 6 in. below height of bottom of hot leg x RVLIS span = 64.5%

C = Height of top of fuel above vessel bottom x RVLIS span = 59.8%

At RNP RVLIS is normally not available when in Mode 6. The RVLIS connection to the reactor vessel head is removed prior to removing the reactor vessel head. Under those conditions where RCS level cannot be monitored, classification should be made based on CS1.3.

NEI 99-01 Basis:

This IC addresses a significant and prolonged loss of ~~(reactor vessel/RCS- [PWR] or RPV [BWR])~~ inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RCS/~~reactor vessel~~ level cannot be restored, fuel damage is probable.

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS/~~reactor vessel~~ levels of ~~EALs 1.b~~CS1.1 and ~~2.b~~CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

~~In EAL 3.a, the 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncover has actually occurred (i.e., to~~

ATTACHMENT 1

EAL Bases

~~account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.~~

~~The inability to monitor (reactor vessel/RCS [PWR] or RPV [BWR]) level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the (reactor vessel/RCS [PWR] or RPV [BWR]).~~

~~These~~ This EALs 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 AG4RG1

RNP Basis Reference(s):

1. GP-008, Draining the Reactor Coolant System
2. UFSAR Table 5.3.0-1
3. UFSAR Figure 5.3.0-1
4. NEI 99-01 CS1

ATTACHMENT 1

EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory affecting core decay heat removal capability

EAL:

CS1.3 Site Area Emergency

RCS water level cannot be monitored for ≥ 30 min. (Note 1)

AND

Core uncover is indicated by **any** of the following:

- UNPLANNED increase in **any** Table C-1 sump or tank due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage
- Containment High Range Radiation Monitor R-32A or R-32B > 5 R/hr
- Erratic source range monitor indication

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

| Table C-1 Sumps / Tanks |
|--|
| <ul style="list-style-type: none">• Containment (CV) sumps• PRT• RCDT• CCW surge tank |

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

RNP Basis:

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EAL Bases

In this EAL, all RCS water level indication would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications. Sump level increases must be evaluated against other potential sources of leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Sumps and tanks where RCS leakage may accumulate are listed in Table C-1. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2, 3).

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed area radiation monitors (R-32A or R-32B). Any positive reading on R-32A or R-32B should be considered an indication of core uncover, either due to the shine from the uncovered core, or the initiation of clad damage. Given that the minimum range of the instrument is 1 R/hr and the instrument range is seven decades, 5 R/hr represents the lowest reading that is considered a clear positive response. If these radiation monitors reach and exceed 5 R/hr, a loss of inventory with potential to uncover the core is likely to have occurred (ref. 6).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations (ref. 4, 5).

NEI 99-01 Basis:

This IC addresses a significant and prolonged loss of ~~(reactor vessel/RCS- [PWR] or RPV [BWR])~~ inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RCS/~~reactor vessel~~ level cannot be restored, fuel damage is probable.

~~Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS/reactor vessel levels of EALs 1.b and 2.b reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.~~

~~In EAL 3.a, t~~The 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncover 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.

ATTACHMENT 1

EAL Bases

The inability to monitor ~~(reactor vessel/RCS [PWR] or RPV [BWR])~~ level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the ~~(reactor vessel/RCS [PWR] or RPV [BWR])~~.

~~These~~**This** EALs addresses concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal; SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues; NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States; and NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management.

Escalation of the emergency classification level would be via IC CG1 or ~~AG1~~**RG1**

RNP Basis Reference(s):

1. GP-008, Draining the Reactor Coolant System
2. UFSAR Table 5.3.0-1
3. UFSAR Figure 5.3.0-1
4. UFSAR Section 7.2.1.1.7 Nuclear Instrumentation System
5. OP-002, Nuclear Instrumentation System
6. RNP-M/MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels for H. B. Robinson, Unit No. 2
7. NEI 99-01 CS1

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory affecting fuel clad integrity with containment challenged

EAL:

CG1.1 General Emergency

RCS level < 59.8% RVLIS Full Range for ≥ 30 min. (Note 1)

AND

Any Containment Challenge indication, Table C-2

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is not required.

| Table C-2 Containment Challenge Indications |
|---|
| <ul style="list-style-type: none">• CONTAINMENT CLOSURE not established (Note 6)• Containment hydrogen concentration $\geq 4\%$• UNPLANNED rise in Containment pressure |

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. In accordance with OMM-033, Implementation of CV Closure, Containment closure exits when:

- The equipment hatch is closed and held in place by a sufficient number of bolts as identified in CM-603, AND
- At least one door in the personnel air lock is closed, AND
- Each penetration providing direct access from the Containment atmosphere to the outside atmosphere is either:
 - closed by a manual or automatic isolation valve, blind flange, or equivalent,

ATTACHMENT 1 EAL Bases

OR

- capable of being closed by an OPERABLE Containment Ventilation Isolation System.

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.

RNP Basis:

59.8% RVLIS Full Range corresponds to the top of active fuel. Other RCS level instruments are off-scale low when core uncover occurs (ref. 1, 2).

The RVLIS full range threshold has been determined as follows (ref. 2, 3):

| Component Dimensions | | RVLIS Full Range (%) |
|---|----------|----------------------|
| Reactor Vessel bottom head OD to top of Control Rod Mechanism housing (in.) | 498.000 | NA |
| Thickness of bottom head (in.) | 5.187 | NA |
| Thickness of vessel head (in.) | 7.750 | NA |
| Height of Control Rod Mechanism above vessel closure head (in.) | 18.000 | NA |
| Inner height of vessel (in.): $498.000 - 5.187 - 7.750 - 18.000 =$ | 467.063 | 100.0 |
| Bottom of vessel (in.) | 0.000 | 0.0 |
| RVLIS span %/in.: $(100.0 - 0.0)/(467.063 - 0.000) =$ | 0.214 | NA |
| Height of RCS hot leg centerline above vessel bottom (in.) | 321.7813 | NA |
| RCS hot leg penetration diameter (in.) | 29.000 | NA |
| Height of bottom of RCS hot leg above vessel bottom (in.): $321.783 - (29.000/2) =$ | 307.2813 | A |
| 6 in. below height of bottom of hot leg (in.): $307.2813 - 6 =$ | 301.2813 | B |
| Height of top of fuel above vessel bottom (in.) | 279.5313 | C |

A = Height of bottom of RCS hot leg above vessel bottom x RVLIS span = 65.8%

B = 6 in. below height of bottom of hot leg x RVLIS span = 64.5%

C = Height of top of fuel above vessel bottom x RVLIS span = 59.8%

At RNP RVLIS is normally not available when in Mode 6. The RVLIS connection to the reactor vessel head is removed prior to removing the reactor vessel head. Under those conditions where RCS level cannot be monitored, classification should be made based on CG1.2.

Three conditions are associated with a challenge to containment integrity:

- CONTAINMENT CLOSURE is not established.
- In the early stages of a core uncover event, it is unlikely that hydrogen buildup due to a core uncover could result in an explosive mixture of dissolved gases in the containment. However, containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. An explosive mixture can be formed when hydrogen gas

ATTACHMENT 1 EAL Bases

concentration in the containment atmosphere is greater than or equal to 4% by volume in the presence of oxygen. Two Containment hydrogen concentration monitors (with a range of 0 to 10% hydrogen) are provided on the Core Cooling and Containment Monitor in the Control Room. Hydrogen concentration is also displayed on ERFIS Points SSC-2512A and SSC-2513A.

- Any unplanned increase in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of containment closure capability. Unplanned containment pressure increases indicates containment closure cannot be assured and the containment cannot be relied upon as a barrier to fission product release.

NEI 99-01 Basis:

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/~~reactor vessel~~ level cannot be restored, fuel damage is probable.

With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to Containment integrity.

In the early stages of a core uncover event, it is unlikely that hydrogen buildup due to a core uncover 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, it~~ 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 uncover 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.

ATTACHMENT 1

EAL Bases

~~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]).~~

Thise EALs 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*.

RNP Basis Reference(s):

1. GP-008, Draining the Reactor Coolant System
2. UFSAR Table 5.3.0-1
3. UFSAR Figure 5.3.0-1
4. NEI 99-01 CG1

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory affecting fuel clad integrity with containment challenged

EAL:

CG1.2 General Emergency

RCS level **cannot** be monitored for ≥ 30 min. (Note 1)

AND

Core uncover is indicated by **any** of the following:

- UNPLANNED increase in **any** Table C-1 sump or tank due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage
- Containment High Range Radiation Monitor R-32A or R-32B > 5 R/hr
- Erratic source range monitor indication

AND

Any Containment Challenge indication, Table C-2

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is not required.

| Table C-1 Sumps / Tanks |
|---|
| <ul style="list-style-type: none">• Containment sumps• PRT• RCDT• CCW surge tank |

ATTACHMENT 1
EAL Bases

| Table C-2 Containment Challenge Indications |
|---|
| <ul style="list-style-type: none">• CONTAINMENT CLOSURE not established (Note 6)• Containment hydrogen concentration $\geq 4\%$• UNPLANNED rise in Containment pressure |

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. In accordance with OMM-033, Implementation of CV Closure, Containment closure exits when:

- The equipment hatch is closed and held in place by a sufficient number of bolts as identified in CM-603, AND
- At least one door in the personnel air lock is closed, AND
- Each penetration providing direct access from the Containment atmosphere to the outside atmosphere is either:
 - closed by a manual or automatic isolation valve, blind flange, or equivalent,
 - OR
 - capable of being closed by an OPERABLE Containment Ventilation Isolation System.

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.

RNP Basis:

In this EAL, all RCS water level indication would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications. Sump level increases must be evaluated against other potential sources of leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Sumps and tanks where RCS leakage may accumulate are listed in listed in Table C-1. Visual observation of leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed area radiation monitors (R-32A or R-32B). Any positive reading on R-32A or R-32B should be considered an indication of core uncover, either due to the shine from the uncovered core, or

ATTACHMENT 1

EAL Bases

the initiation of clad damage. Given that the minimum range of the instrument is 1 R/hr and the instrument range is seven decades, 5 Rem/hr represents the lowest reading that is considered a clear positive response. If these radiation monitors reach and exceed 5 R/hr, a loss of inventory with potential to uncover the core is likely to have occurred (ref. 6).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations (ref. 4, 5).

Three conditions are associated with a challenge to containment integrity:

- CONTAINMENT CLOSURE is not established.
- In the early stages of a core uncover event, it is unlikely that hydrogen buildup due to a core uncover could result in an explosive mixture of dissolved gases in the containment. However, containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. An explosive mixture can be formed when hydrogen gas concentration in the containment atmosphere is greater than 4% by volume in the presence of oxygen.
- Any unplanned increase in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of containment closure capability. Unplanned containment pressure increases indicates containment closure cannot be assured and the containment cannot be relied upon as a barrier to fission product release.

NEI 99-01 Basis:

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/~~reactor vessel~~ level cannot be restored, fuel damage is probable.

With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to Containment integrity.

In the early stages of a core uncover event, it is unlikely that hydrogen buildup due to a core uncover could result in an explosive gas mixture in containment. If all installed hydrogen gas

ATTACHMENT 1

EAL Bases

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, the~~ 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 uncover 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])~~.

~~This~~ ~~ese~~ EALs ~~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*.

RNP Basis Reference(s):

1. GP-008, Draining the Reactor Coolant System
2. UFSAR Table 5.3.0-1
3. UFSAR Figure 5.3.0-1
4. UFSAR Section 7.2.1.1.7 Nuclear Instrumentation System
5. OP-002, Nuclear Instrumentation System
6. RNP-M/MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels for H. B. Robinson, Unit No. 2
7. NEI 99-01 CG1

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

Initiating Condition: Loss of **all but one** AC power source to emergency buses for 15 minutes or longer

EAL:

CU2.1 Unusual Event

AC power capability to 480V emergency buses E-1 and E-2 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 Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling, D - Defueled

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.

RNP Basis:

Emergency buses E-1 and E-2 are the essential buses.

For emergency classification purposes, “capability” means that an AC power source is available to the emergency buses, whether or not the buses are powered from it.

ATTACHMENT 1 EAL Bases

Offsite power is connected to the 230KV switchyard through six lines:

- Darlington County Plant south through 230 KV CB 52/11 and 230 KV CB 52/12
- Darlington County Plant north through 230 KV CB 52/13 and 230 KV CB 52/14
- Sumter through 230 KV CB 52/10 and 230 KV CB 52/11
- Rockingham through 230 KV CB 52/2 and 230 KV CB 52/3
- Darlington SCPSA through 230 KV CB 52/7 and 230 KV CB 52/8
- Florence through 230 KV CB 52/4 and 230 KV CB 52/5

The Unit 1 115KV switchyard is connected to the 230KV through two lines:

- #1 Auto transformer to 115 KV west bus through 230 KV CBs 52/5 and 52/6
- #2 Auto transformer to 115 KV east bus through 230 KV CBs 52/1 and 52/2

Emergency bus E-1 is supplied from 4160V bus 2 through Station Service Transformer 2F (SST 2F). When the Main Generator is shutdown, 4160V bus 2 is supplied from the Startup Transformer. Emergency bus E-2 is supplied from 4160V bus 3 through Station Service Transformer 2G (SST 2G). 4160V bus 3 is normally supplied from the Startup Transformer. The first source of offsite emergency power is the 115 KV to 4160V Startup Transformer. This transformer is supplied from the lines connecting to the 115KV grid and two ties to the 230KV grid.

Another method to obtain offsite power is by backfeeding the emergency buses through the Main and Unit Auxiliary Transformer. This is only done during Cold Shutdown unless nuclear safety considerations require it to be done during Hot Shutdown when no other power sources are available. Buses E-1 and E-2 can also be supplied from onsite Emergency Diesel Generators (EDGs) A and B, respectively (ref.1, 2, 3, 4, 5, 6).

This cold condition EAL is equivalent to the hot condition EAL SA1.1.

NEI 99-01 Basis:

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as an Alert because of the increased time available to restore another power source to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an emergency bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power

ATTACHMENT 1

EAL Bases

source (e.g., an onsite diesel generator).

- A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from the unit main generator.
- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being ~~back~~-fed from an offsite power source.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

The subsequent loss of the remaining single power source would escalate the event to an Alert in accordance with IC CA2.

RNP Basis Reference(s):

1. OP-603, Electrical Distribution
2. UFSAR Figure 8.1.2-1a
3. UFSAR Section 8.3 Onsite Power Systems
4. AOP-024, Loss of Instrument Bus
5. DBD/R87038/SD16, Electrical Distribution System
6. EOP-ECA-0.0, Loss of All AC Power
7. NEI 99-01 CU2

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite and **all** onsite AC power to emergency buses for 15 minutes or longer

EAL:

CA2.1 Alert

Loss of **all** offsite and **all** onsite AC power capability to 480V emergency buses E-1 and E-2 for ≥ 15 min. (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, D - Defueled

Definition(s):

None

RNP Basis:

Emergency buses E-1 and E-2 are the essential buses.

For emergency classification purposes, “capability” means that an AC power source is available to the emergency buses, whether or not the buses are powered from it.

Offsite power is connected to the 230KV switchyard through six lines:

- Darlington County Plant south through 230 KV CB 52/11 and 230 KV CB 52/12
- Darlington County Plant north through 230 KV CB 52/13 and 230 KV CB 52/14
- Sumter through 230 KV CB 52/10 and 230 KV CB 52/11
- Rockingham through 230 KV CB 52/2 and 230 KV CB 52/3
- Darlington SCPSA through 230 KV CB 52/7 and 230 KV CB 52/8
- Florence through 230 KV CB 52/4 and 230 KV CB 52/5

The Unit 1 115KV switchyard is connected to the 230KV through two lines:

- #1 Auto transformer to 115 KV west bus through 230 KV CBs 52/5 and 52/6
- #2 Auto transformer to 115 KV east bus through 230 KV CBs 52/1 and 52/2

ATTACHMENT 1

EAL Bases

Emergency bus E-1 is supplied from 4160V bus 2 through Station Service Transformer 2F (SST 2F). When the Main Generator is shutdown, 4160V bus 2 is supplied from the Startup Transformer. Emergency bus E-2 is supplied from 4160V bus 3 through Station Service Transformer 2G (SST 2G). 4160V bus 3 is normally supplied from the Startup Transformer. The first source of offsite emergency power is the 115 KV to 4160V Startup Transformer. This transformer is supplied from the lines connecting to the 115KV grid and two ties to the 230KV grid

Another method to obtain offsite power is by backfeeding the emergency buses through the Main and Unit Auxiliary Transformer. This is only done during Cold Shutdown unless nuclear safety considerations require it to be done during Hot Shutdown when no other power sources are available. Buses E-1 and E-2 can also be supplied from onsite Emergency Diesel Generators (EDGs) A and B, respectively (ref.1, 2, 3, 4, 5, 6).

This cold condition EAL is equivalent to the hot condition loss of all offsite AC power EAL SS1.1.

NEI 99-01 Basis:

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as a Site Area Emergency because of the increased time available to restore an emergency bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition represents an actual or potential substantial degradation of the level of safety of the plant.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via IC CS1 or **AS4RS1**.

RNP Basis Reference(s):

1. OP-603, Electrical Distribution
2. UFSAR Figure 8.1.2-1a
3. UFSAR Section 8.3 Onsite Power Systems
4. AOP-024, Loss of Instrument Bus
5. DBD/R87038/SD16, Electrical Distribution System
6. EOP-ECA-0.0, Loss of All AC Power
7. NEI 99-01 CA2

ATTACHMENT 1

EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.1 Unusual Event

UNPLANNED increase in RCS temperature to > 200°F due to loss of decay heat removal capability

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

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.

RNP Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specifications cold shutdown temperature limit (200°F, ref. 1). These include Core Exit Thermocouples (CETs) and the wide range (0-700°F) RTDs located in the hot and cold legs of the RCS:

| | <u>Cold Leg</u> | <u>Hot Leg</u> |
|--------|-----------------|-----------------------|
| Loop 1 | TE-410 | TE-413-1 and TE-413-2 |
| Loop 2 | TE-420 | TE-423 |
| Loop 3 | TE-430 | TE-433 |

TE-413 is a dual element RTD with TE-413-1 providing indication to TR-413, and TE-413-2 providing indication to the Inadequate Core Cooling Monitor (ICCM). Temperatures are also recorded on TR-413 (T_{hot}) and TR-410 (T_{cold}), which are located on the RTGB. The temperatures of the hot and cold legs can also be read on the core subcooling monitor (designated T_h 1, T_h 2, T_h 3, T_c 1, T_c 2, and T_c 3) and are used for indication during heatup and cooldown. RCS/RHR pump discharge temperature indication, such as TR-604, can also be used to monitor RCS temperature (ref. 2, 3).

In the absence of reliable RCS temperature indication caused by a loss of decay heat removal capability, classification should be based on EAL CU3.2 should RCS level indication be subsequently lost.

ATTACHMENT 1

EAL Bases

NEI 99-01 Basis:

This IC addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit, ~~or the inability to determine RCS temperature and level, and~~ represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the ~~Emergency Director~~ Emergency Coordinator should also refer to IC CA3.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

~~EAL #1~~ This EAL involves a loss of decay heat removal capability, or an addition of heat to the RCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

During an outage, the level in the reactor vessel will normally be maintained ~~at or~~ above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid increase in reactor coolant temperature depending on the time after shutdown.

~~—— EAL #2 reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.~~

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

RNP Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. GP-002, Cold Shutdown to Hot Subcritical at No-Load T-AVG
3. GP-007, Plant Cooldown from Hot Shutdown to Cold Shutdown Conditions
4. NEI 99-01 CU3

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

| |
|---------------------------------|
| CU3.2 Unusual Event |
|---------------------------------|

| |
|---|
| Loss of all RCS temperature and RCS level indication for ≥ 15 min. (Note 1) |
|---|

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6- Refueling

Definition(s):

None

RNP Basis:

Reactor vessel water level is normally monitored using the following instruments (ref. 2):

- LI-403
- LI-404
- Standpipe Loop B
- Standpipe Loop C
- RVLIS (LT-511AB and LT-511BB)
- Remote camera, if vessel head is removed

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specifications cold shutdown temperature limit of 200°F (ref. 1). These include Core Exit Thermocouples (CETs) and the wide range (0-700°F) RTDs located in the hot and cold legs of the RCS:

ATTACHMENT 1 EAL Bases

| <u>Cold Leg</u> | <u>Hot Leg</u> | |
|-----------------|----------------|-----------------------|
| Loop 1 | TE-410 | TE-413-1 and TE-413-2 |
| Loop 2 | TE-420 | TE-423 |
| Loop 3 | TE-430 | TE-433 |

TE-413 is a dual element RTD with TE-413-1 providing indication to TR-413, and TE-413-2 providing indication to the Inadequate Core Cooling Monitor (ICCM). Temperatures are also recorded on TR-413 (T_{hot}) and TR-410 (T_{cold}), which are located on the RTGB. The temperatures of the hot and cold legs can also be read on the core subcooling monitor (designated T_h 1, T_h 2, T_h 3, T_c 1, T_c 2, and T_c 3) and are used for indication during heatup and cooldown. RCS/RHR pump discharge temperature indication, such as TR-604, can also be used to monitor RCS temperature (ref. 3, 4).

NEI 99-01 Basis:

This ~~IC-EAL~~ addresses ~~an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit, or~~ the inability to determine RCS temperature and level, ~~and~~ represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the ~~Emergency Director~~Emergency Coordinator 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.

ATTACHMENT 1
EAL Bases

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

RNP Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. GP-008, Draining the Reactor Coolant System
3. GP-002, Cold Shutdown to Hot Subcritical at No-Load T-AVG
4. GP-007, Plant Cooldown from Hot Shutdown to Cold Shutdown Conditions
5. NEI 99-01 CU3

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.1 Alert

UNPLANNED increase in RCS temperature to > 200°F for > Table C-3 duration
(Note 1)

OR

UNPLANNED RCS pressure increase > 10 psig due to a loss of RCS cooling (this does **not** apply during water-solid plant conditions)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

| Table C-3: RCS Heat-up Duration Thresholds | | |
|--|----------------------------|------------------|
| RCS Status | Containment Closure Status | Heat-up Duration |
| Intact (but not REDUCED INVENTORY) | N/A | 60 min.* |
| Not intact OR At REDUCED INVENTORY | established | 20 min.* |
| | not established | 0 min. |
| * If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable. | | |

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to secure Containment as a functional barrier to fission product release during plant shutdown conditions. In accordance with OMM-033, Implementation of CV Closure, Containment closure exits when:

- The equipment hatch is closed and held in place by a sufficient number of bolts as identified in CM-603, AND
- At least one door in the personnel air lock is closed, AND
- Each penetration providing direct access from the Containment atmosphere to the outside atmosphere is either:

ATTACHMENT 1

EAL Bases

- closed by a manual or automatic isolation valve, blind flange, or equivalent,
OR
- capable of being closed by an OPERABLE Containment Ventilation Isolation System.

REDUCED INVENTORY - Plant condition when fuel is in the Reactor Vessel and Reactor Coolant System level is less than or equal to -36 inches below the vessel flange.

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.

RNP Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specifications cold shutdown temperature limit of 200°F (ref. 1). These include the wide range (0-700°F) RTDs located in the hot and cold legs of the RCS:

| <u>Cold Leg</u> | <u>Hot Leg</u> | |
|-----------------|----------------|-----------------------|
| Loop 1 | TE-410 | TE-413-1 and TE-413-2 |
| Loop 2 | TE-420 | TE-423 |
| Loop 3 | TE-430 | TE-433 |

TE-413 is a dual element RTD with TE-413-1 providing indication to TR-413, and TE-413-2 providing indication to the Inadequate Core Cooling Monitor (ICCM). Temperatures are also recorded on TR-413 (T_{hot}) and TR-410 (T_{cold}), which are located on the RTGB. The temperatures of the hot and cold legs can also be read on the core subcooling monitor (designated $T_h 1$, $T_h 2$, $T_h 3$, $T_c 1$, $T_c 2$, and $T_c 3$) and are used for indication during heatup and cooldown. RCS/RHR pump discharge temperature indication, such as TR-604, can also be used to monitor RCS temperature (ref. 2, 3).

PI-403, RCS Narrow Range Pressure (0-1000 psi), is graduated in 20 psi increments and is capable of measuring pressure to less than 10 psig (ref. 2).

In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the RCS pressure increase criteria when in Mode 5 or based on time to boil data when in Mode 6.

ATTACHMENT 1

EAL Bases

NEI 99-01 Basis:

This IC addresses conditions involving a loss of decay heat removal capability or an addition of heat to the RCS in excess of that which can currently be removed. Either condition represents an actual or potential substantial degradation of the level of safety of the plant.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

The RCS Heat-up Duration Thresholds table addresses an increase in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not intact, or RCS inventory is reduced (e.g., mid-loop operation ~~in PWRs~~). The 20-minute criterion was included to allow time for operator action to address the temperature increase.

The RCS Heat-up Duration Thresholds table also addresses an increase in RCS temperature with the RCS intact. The status of CONTAINMENT CLOSURE is not crucial in this condition since the intact RCS is providing a high pressure barrier to a fission product release. The 60-minute time frame should allow sufficient time to address the temperature increase without a substantial degradation in plant safety.

Finally, in the case where there is an increase in RCS temperature, the RCS is not intact or is at reduced inventory ~~[PWR]~~, and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because 1) the evaporated reactor coolant may be released directly into the containment atmosphere and subsequently to the environment, and 2) there is reduced reactor coolant inventory above the top of irradiated fuel.

~~EAL #2~~The RCS pressure increase threshold provides a pressure-based indication of RCS heat-up in the absence of RCS temperature monitoring capability.

Escalation of the emergency classification level would be via IC CS1 or ~~AS1~~RS1.

RNP Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. GP-002, Cold Shutdown to Hot Subcritical at No-Load T-AVG
3. GP-007, Plant Cooldown from Hot Shutdown to Cold Shutdown Conditions
6. NEI 99-01 CA3

ATTACHMENT 1

EAL Bases

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

< 109.5 VDC (Bus A) / < 106.2 (Bus B) bus voltage indications on Technical Specification required 125 VDC buses for ≥ 15 min. (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

None

RNP Basis:

The A and B batteries are safety-related and are equipped with two redundant battery chargers per bus. The A and B batteries are sized to carry expected shutdown loads following a design basis accident with no battery chargers available for a period of 1 hour without battery terminal voltage falling below minimum allowable voltage. The four safety-related chargers are sized to charge a partially discharged battery within 24 hours while carrying its normal load (ref. 1).

Minimum battery terminal voltage is 1.75 VDC per cell for each of 60 cells per battery or 105 VDC (ref. 1). Calculations performed for the B battery replacement, however, specify minimum battery terminal voltage of 106.8 VDC and a corresponding bus voltage of 106.2 VDC (ref. 2). Battery A minimum bus voltage was calculated to be 109.5 VDC (ref. 11). Control Room annunciator APP-036-D3, BATT A/B LO VOLT, is received at 123 VDC and signals sustained loss of a battery charger or battery/cell failure (ref. 3). Battery bus voltage is indicated on ERFIS Points APV3022A (MCC-A) and APV3023A (MCC-B) (ref. 4).

This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS7.1.

ATTACHMENT 1

EAL Bases

NEI 99-01 Basis

This IC addresses a loss of vital DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or refueling mode. In these modes, the core decay heat load has been significantly reduced, and coolant system temperatures and pressures are lower; these conditions increase the time available to restore a vital DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, “required” means the vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if Train A is out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an Unusual Event. A loss of Vital DC power to Train A would not warrant an emergency classification.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Depending upon the event, escalation of the emergency classification level would be via IC CA1 or CA3, or an IC in Recognition Category **AR**.

RNP Basis Reference(s):

1. UFSAR Section 8.3.2 DC Power System (125 Volt)
2. Calculation RNP-E-6.031, Station Battery B Replacement, Figure 1
3. APP-036-D3 BATT A/B LO VOLT
4. UFSAR Figure 8.3.1-5
5. Technical Specifications 3.8.4, DC Sources - Operating
6. Technical Specifications 3.8.5, DC Sources - Shutdown
7. Technical Specifications 3.8.6, Battery Cell Parameters
8. OP-601, DC Supply System
9. EPP-26, Loss of DC Bus A
10. EPP-27, Loss of DC Bus B
11. Calculation RNP-E-6.018, Section 5.1.4
12. NEI 99-01 CU4

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

CU5.1 Unusual Event

Loss of **all** Table C-4 onsite communication methods

OR

Loss of **all** Table C-4 offsite communication methods

OR

Loss of **all** Table C-4 NRC communication methods

| Table C-4 Communication Methods | | | |
|--|---------------|----------------|------------|
| System | Onsite | Offsite | NRC |
| Public Address System | X | | |
| PBX Telephone System | X | | |
| Radio Transceivers for RNP and Vicinity | X | | |
| Back-up Telephone System (ESSX) | X | | |
| Plant Security Radio Transceivers | X | | |
| Corporate Telephone Communications System (Voicenet) | | X | X |
| BellSouth | | X | X |
| Dedicated Telephone System to Load Dispatcher | | X | |
| Plant Security Radio Control Station | | X | |
| DEMNET | | X | |
| NRC Emergency Telecommunication System (ETS) | | | X |
| Satellite Phones | | X | X |
| Cellular Phones | | X | X |
| Palmetto 800 Transceivers | | X | |

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, D – Defueled

Definition(s):

ATTACHMENT 1 EAL Bases

None

RNP Basis:

Onsite, offsite and NRC communications include one or more of the systems listed in Table C-4 (ref. 1, 2).

The NRC ETS Phone and the NRC HPN Phone are part of the PABX and will be unavailable if the PABX is unavailable.

This EAL is the cold condition equivalent of the hot condition EAL SU7.1.

NEI 99-01 Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to **Offsite Response Organizations (OROs)** and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

EAL #1The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

EAL #2The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are ~~(see Developer Notes)~~ the State, Darlington, Lee and Chesterfield County EOCs.

EAL #3The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

RNP Basis Reference(s):

1. PLP-007, Emergency Plan, Attachment 6.1
2. UFSAR Section 9.5.2 Communications Systems
3. NEI 99-01 CU5

ATTACHMENT 1

EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting Safety Systems

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

EAL:

CA6.1 Alert

The occurrence of **any** Table C-5 hazardous event

AND EITHER:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused **VISIBLE DAMAGE** to a SAFETY SYSTEM component or structure needed for the current operating mode

Table C-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:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

ATTACHMENT 1

EAL Bases

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

- (1) The integrity of the reactor coolant pressure boundary;
- (2) The capability to shut down the reactor and maintain it in a safe shutdown condition;
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

VISIBLE DAMAGE - Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

RNP Basis:

- The significance of seismic events are discussed under EAL HU2.1 (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2).
- The plant Seismic Category I structures are designed to withstand the effects of the design wind, 83 mph (108 gust). (ref. 3, 4).
- An explosion that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

ATTACHMENT 1
EAL Bases

NEI 99-01 Basis:

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

~~EAL-1.b.1~~ **The first conditional** addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

~~EAL-1.b.2~~ **The second conditional** addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC CS1 or **AS4RS1**.

RNP Basis Reference(s):

1. AOP-021 Seismic Disturbances
2. RNP-F/PSA-0009, Assessment of Internal Flooding Events
3. UFSAR Table 3.3.1-1
4. OMM-021, Operation During Adverse Weather Conditions
5. NEI 99-01 CA6

ATTACHMENT 1
EAL Bases

Category H – Hazards and Other Conditions Affecting Plant Safety

EAL Group: ANY (EALs in this category are applicable to any plant condition, hot or cold.)

Hazards are non-plant, system-related events that can directly or indirectly affect plant operation, reactor plant safety or personnel safety.

1. Security

Unauthorized entry attempts into the Protected Area, bomb threats, sabotage attempts, and actual security compromises threatening loss of physical control of the plant.

2. Seismic Event

Natural events such as earthquakes have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety.

3. Natural or Technology Hazard

Other natural and non-naturally occurring events that can cause damage to plant facilities include tornados, FLOODING, hazardous material releases and events restricting site access warranting classification.

4. Fire

Fires can pose significant hazards to personnel and reactor safety. Appropriate for classification are fires within the site Protected Area or 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.

ATTACHMENT 1 EAL Bases

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 Coordinator 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 Coordinator the latitude to classify emergency conditions consistent with the established classification criteria based upon Emergency Coordinator judgment.

ATTACHMENT 1

EAL Bases

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1 Unusual Event

A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift 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):

SECURITY CONDITION - Any security event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A security condition does not involve a hostile action.

HOSTILE ACTION - An act toward RNP 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 RNP. 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).

RNP Basis:

This EAL is based on the RNP Security Plan (ref. 1).

Reports from Security Shift Supervision may be made via non-supervisory security personnel such as the CAS operator.

ATTACHMENT 1

EAL Bases

NEI 99-01 Basis:

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1, 2). Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

~~EAL #1~~The first threshold references ~~(site-specific-the security~~ Security Shift Supervision~~shift supervision)~~ because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.39 information.

~~EAL #2~~The second threshold addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with ~~(site-specific-procedure)~~the RNP Security Plan and DBT (ref. 1).

~~EAL #3~~The third threshold addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with the RNP Security Plan and DBT (ref. 1)~~(site-specific-procedure)~~.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the RNP Security Plan (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

RNP Basis Reference(s):

1. RNP Security Plan and DBT
2. AOP-034, Security Events
3. NEI 99-01 HU1

ATTACHMENT 1

EAL Bases

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Hostile action within the owner controlled area or airborne attack threat

EAL:

HA1.1 Alert

A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the 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):

HOSTILE ACTION - An act toward RNP 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 RNP. 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 - That area surrounding the Protected Area beyond which RNP exercises access control.

RNP Basis:

Reports from Security Shift Supervision may be made via non-supervisory security personal such as the CAS operator.

NEI 99-01 Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between the Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1, 2).

ATTACHMENT 1

EAL Bases

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan ~~and Independent Spent Fuel Storage Installation Security Program~~*.

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of Offsite Response Organizations (OROs), allowing them to be better prepared should it be necessary to consider further actions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

~~EAL #1~~The first threshold is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes any action directed against an ISFSI that is located outside the plant PROTECTED AREA.

~~EAL #2~~The second threshold addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and OROs are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with ~~{site-specific security procedures}~~.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may be provided by NORAD through the NRC.

In some cases, it may not be readily apparent if an aircraft impact within the OWNER CONTROLLED AREA was intentional (i.e., a HOSTILE ACTION). It is expected, although not certain, that notification by an appropriate Federal agency to the site would clarify this point. In this case, the appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. The emergency declaration, including one based on other ICs/EALs, should not be unduly delayed while awaiting notification by a Federal agency.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the ~~RNP Security Plan and DBT (ref. 1)~~.

RNP Basis Reference(s):

1. RNP Security Plan and DBT
2. AOP-034, Security Events
3. NEI 99-01 HA1

ATTACHMENT 1

EAL Bases

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Hostile Action within the Protected Area

EAL:

HS1.1 Site Area Emergency

A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward RNP 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 RNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated Security area around the process buildings and is depicted in Drawing HBR2-9800, Plot Plan RNP.

RNP Basis:

Reports from Security Shift Supervision may be made via non-supervisory security personal such as the CAS operator.

NEI 99-01 Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the PROTECTED AREA. This event will require rapid response and assistance due to the possibility for damage to plant equipment.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1, 2).

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~~*.

ATTACHMENT 1

EAL Bases

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Site Area Emergency declaration will mobilize **Offsite Response Organization (ORO)** resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

This IC does not apply to a HOSTILE ACTION directed at an ISFSI PROTECTED AREA located outside the plant PROTECTED AREA; such an attack should be assessed using IC HA1. It also does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the **RNP Security Plan and DBT (ref. 1)**.

Escalation of the emergency classification level would be via IC HG1.

RNP Basis Reference(s):

1. RNP Security Plan and DBT
2. AOP-034, Security Events
3. NEI 99-01 HS1

ATTACHMENT 1

EAL Bases

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Hostile Action resulting in loss of physical control of the facility

EAL:

HG1.1 General Emergency

A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision

AND EITHER of the following has occurred:

Any of the following safety functions cannot be controlled or maintained

- Reactivity
- Core cooling
- RCS heat removal

OR

Damage to spent fuel has occurred or is IMMINENT

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward RNP 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 RNP. 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

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated Security area around the process buildings and is depicted in Drawing HBR2-9800, Plot Plan RNP.

ATTACHMENT 1

EAL Bases

RNP Basis:

Reports from Security Shift Supervision may be made via non-supervisory security personnel such as the CAS operator.

NEI 99-01 Basis:

This IC addresses an event in which a HOSTILE FORCE has taken physical control of the facility to the extent that the plant staff can no longer operate equipment necessary to maintain key safety functions. It also addresses a HOSTILE ACTION leading to a loss of physical control that results in actual or IMMINENT damage to spent fuel due to 1) damage to a spent fuel pool cooling system (e.g., pumps, heat exchangers, controls, etc.) or, 2) loss of spent fuel pool integrity such that sufficient water level cannot be maintained.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1, 2).

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan ~~and Independent Spent Fuel Storage Installation Security Program~~*.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the **RNP Security Plan and DBT (ref.1)**.

RNP Basis Reference(s):

1. RNP Security Plan and DBT
2. AOP-034, Security Events
3. NEI 99-01 HG1

ATTACHMENT 1

EAL Bases

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 Recording Unit A or B indicates seismic event > Operating Basis Earthquake (0.1g horizontal **OR** 0.067g vertical)

Mode Applicability:

All

Definition(s):

None

RNP Basis:

AOP-021 Seismic Disturbances provides the guidance for determining if the OBE earthquake threshold is exceeded and any required response actions (ref. 1).

The Operating Basis Earthquake (OBE) is defined as that earthquake which could reasonably be expected to affect the plant site during the operating life of the plant, based on the earthquake potential of the geographic area. At Robinson Plant, this is defined as half of the vibration defined for an SSE or 0.1g horizontal or 0.067g vertical. Facility design ensures that all equipment necessary to operate the plant without undue risk to the health and safety of the public will remain functional for any seismic event where ground motion is less than that of the OBE (ref. 1, 2).

If the OBE ALARM on Seismic Monitor "B" is illuminated, then the earthquake is greater than an Operating Basis Earthquake (0.1g Horizontal or 0.067g Vertical). If the DBE/SSE ALARM on Seismic Monitor "A" is illuminated, then the earthquake is greater than the Design Basis/Safe Shutdown Earthquake (0.2g Horizontal OR 0.133g Vertical) (ref. 1).

To avoid inappropriate emergency classification resulting from spurious actuation of the seismic instrumentation or felt motion not attributable to seismic activity, an offsite agency (USGS, National Earthquake Information Center) can confirm that an earthquake has occurred in the area of the plant. Such confirmation should not, however, preclude a timely emergency declaration based on receipt of the OBE alarm. When calling the NEIC, select **option #1** and inform the analyst you wish to confirm recent seismic activity in the vicinity of RNP. Provide the analyst with the following RNP coordinates: **34° 24.2' north latitude, 80° 09.5' west longitude** (ref. 3). Alternatively, near real-time seismic activity can be accessed via the NEIC website:

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EAL Bases

<http://earthquake.usgs.gov/eqcenter/>

NEI 99-01 Basis:

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE). An earthquake greater than an OBE but less than a Safe Shutdown Earthquake (SSE) should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., performs walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event (e.g., lateral accelerations in excess of 0.08g). The Shift Manager or Emergency ~~Director~~ Coordinator may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

RNP Basis Reference(s):

1. AOP-21 Seismic Disturbances
2. UFSAR 3.7.4 Seismic Instrumentation
3. UFSAR 1.2.1 Site and Environment
4. NEI 99-01 HU2

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EAL Bases

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 - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated Security area around the process buildings and is depicted in Drawing HBR2-9800, Plot Plan RNP.

RNP Basis:

Response actions associated with a tornado onsite are provided in OMM-021, Operation During Adverse Weather Conditions (ref. 1).

If damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under EAL CA6.1 or SA9.1.

A tornado striking (touching down) within the PROTECTED AREA warrants declaration of an Unusual Event regardless of the measured wind speed at the meteorological tower. A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

NEI 99-01 Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

~~EAL #1~~ EAL HU3.1 addresses a tornado striking (touching down) within the PROTECTED AREA.

~~EAL #2 addresses flooding of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.~~

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EAL Bases

~~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 **AR**, F, S or C.

RNP Basis Reference(s):

1. OMM-021, Operation During Adverse Weather Conditions
2. NEI 99-01 HU3

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EAL Bases

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 needed for the current operating mode

Mode Applicability:

All

Definition(s):

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

- (1) The integrity of the reactor coolant pressure boundary;
- (2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition;
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

RNP Basis:

The primary internal flooding area of concern is the Auxiliary Building (includes the Component Cooling Water Pump Room). Flooding in this area could have the potential to cause a reactor trip and could result in consequential failures to important systems. The potential for flooding in this area was determined by an examination of piping systems in the area and also considered propagation of water from one area to another. The most important internal flooding initiating events are associated with the failure of large service water pipes located on elevation 226 in the Auxiliary Building. Elevation 226 is mostly open to flood propagation, and significant safety-related equipment could be affected by a very large flood that would allow water to accumulate to significant depths. When water level reaches a certain height in the Auxiliary Building, both trains of safeguards equipment can be rendered inoperable. This event

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is further compounded by the fact that all spilled water may become contaminated and must be treated so until proven otherwise. Other flood initiating events on elevation 226 are less important but do have the potential to affect safe plant operations if timely mitigation does not occur to terminate flooding (Ref. 1, 2).

NEI 99-01 Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

~~EAL #1 addresses a tornado striking (touching down) within the PROTECTED AREA.~~

This EAL addresses FLOODING of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

~~EAL #3 addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.~~

~~EAL #4 addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site flooding caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.~~

~~This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.~~

~~EAL #5 addresses (site-specific description).~~

Escalation of the emergency classification level would be based on ICs in Recognition Categories AR, F, S or C.

RNP Basis Reference(s):

1. RNP-F/PSA-0009, Assessment of Internal Flooding Events
2. RSC 99-17, RNP Probabilistic Safety Assessment, Section 3.5, Internal Flooding Initiating Events, Table 3.12
3. AOP-022, Loss of Service Water
4. AOP-032, Response to Flooding from the Fire Protection System
5. NEI 99-01 HU3

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EAL Bases

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 offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

Mode Applicability:

All

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated Security area around the process buildings and is depicted in Drawing HBR2-9800, Plot Plan RNP.

RNP Basis:

As used here, the term "offsite" is meant to be areas external to the RNP PROTECTED AREA.

NEI 99-01 Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

~~EAL #1 addresses a tornado striking (touching down) within the PROTECTED AREA.~~

~~This EAL addresses flooding of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.~~

~~EAL #3~~ This EAL addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.

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~~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 **AR**, F, S or C.

RNP Basis Reference(s):

1. NEI 99-01 HU3

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EAL Bases

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):

None

RNP Basis:

None

NEI 99-01 Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant. ~~EAL #1 addresses a tornado striking (touching down) within the PROTECTED AREA.~~

~~This EAL addresses flooding of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.~~

~~EAL #3 addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.~~

~~EAL #4~~ This EAL addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site FLOODING caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.

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EAL Bases

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 **AR**, F, S or C.

RNP Basis Reference(s):

1. NEI 99-01 HU3

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EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.1 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 Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas

- Containment
- Auxiliary Building
- Control Room
- Fuel Handling Building
- Intake
- AFW Room
- 4 KV Switchgear Room
- E-1/E-2 Switchgear Room
- RWST
- CST

Mode Applicability:

All

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EAL Bases

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.

RNP Basis:

The 15 minute requirement begins with a credible notification that a fire is occurring, or receipt of multiple valid fire detection system alarms or field validation of a single fire alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field.

Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1, 2).

NEI 99-01 Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

EAL #1

~~The~~ For EAL HU4.1 the intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report. EAL #2

~~This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30 minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.~~

~~A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.~~

~~If an actual FIRE is verified by a report from the field, then EAL #1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15 minutes of the~~

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~~report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30 minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.~~

~~EAL #3~~

~~In addition to a FIRE addressed by EAL #1 or EAL #2, a FIRE within the plant PROTECTED AREA not extinguished within 60 minutes may also potentially degrade the level of plant safety. This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA. [Sentence for plants with an ISFSI outside the plant Protected Area]~~

~~EAL #4~~

~~If a FIRE within the plant or ISFSI [for plants with an ISFSI outside the plant Protected Area] PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.~~

~~Basis-Related Requirements from Appendix R~~

~~Appendix R to 10 CFR 50, states in part:~~

~~Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."~~

~~When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.~~

~~Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.~~

~~In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL #2, the 30 minutes to verify a single alarm is well within this worst-case 1-hour time period.~~

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

RNP Basis Reference(s):

1. OMP-003, Shutdown Safety Function Guidelines

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2. OMM-003, Fire Protection Pre-Plans/Unit No. 2
3. NEI 99-01 HU4

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EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.2 Unusual Event

Receipt of a single fire alarm (i.e., no other indications of a FIRE)

AND

The fire alarm is indicating a FIRE within **any** Table H-1 area

AND

The existence of a FIRE is not verified within 30 min. of alarm receipt (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

**Table H-1
Fire Areas**

- Containment
- Auxiliary Building
- Control Room
- Fuel Handling Building
- Intake
- AFW Room
- 4 KV Switchgear Room
- E-1/E-2 Switchgear Room
- RWST
- CST

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.

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RNP Basis:

The 30 minute requirement begins upon receipt of a single valid fire detection system alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field. Actual field reports must be made within the 30 minute time limit or a classification must be made. If a fire is verified to be occurring by field report, classification shall be made based on EAL HU4.1.

Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1, 2).

NEI 99-01 Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

EAL #1

~~The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.~~

~~Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.~~

EAL #2

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then ~~EAL #1~~HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

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EAL #3

~~In addition to a FIRE addressed by EAL #1 or EAL #2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety. This basis extends to a FIRE occurring within the PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA. [Sentence for plants with an ISFSI outside the plant Protected Area]~~

EAL #4

Basis-Related Requirements from Appendix R

Appendix R to 10 CFR 50, states in part:

Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."

When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.

Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in this EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

RNP Basis Reference(s):

1. OMP-003, Shutdown Safety Function Guidelines
2. OMM-003, Fire Protection Pre-Plans/Unit No. 2
3. NEI 99-01 HU4

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EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.3 Unusual Event

A FIRE within the plant PROTECTED AREA not extinguished within 60 min. of the initial report, alarm or indication (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

All

Definition(s):

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated Security area around the process buildings and is depicted in Drawing HBR2-9800, Plot Plan RNP.

RNP Basis:

None

NEI 99-01 Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

~~EAL #1~~

~~The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.~~

~~Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a~~

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EAL Bases

~~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 #1HU4.1 or EAL #2HU4.2, 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.~~

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~~Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.~~

~~In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.~~

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

RNP Basis Reference(s):

1. NEI 99-01 HU4

ATTACHMENT 1

EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.4 Unusual Event

A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

Mode Applicability:

All

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 - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated Security area around the process buildings and is depicted in Drawing HBR2-9800, Plot Plan RNP.

RNP Basis:

None

NEI 99-01 Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

EAL #1

~~The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.~~

~~Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.~~

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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.~~

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EAL Bases

~~Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.~~

~~In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in EAL #2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.~~

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

RNP Basis Reference(s):

1. NEI 99-01 HU4

ATTACHMENT 1
EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gases

Initiating Condition: Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

HA5.1 Alert

Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table R-2/H-2 rooms or areas

AND

Entry into the room or area is prohibited or IMPEDED (Note 5)

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

| Table R-2/H-2 Safe Operation & Shutdown Rooms/Areas | |
|---|----------------|
| Room/Area | Mode(s) |
| Reactor Auxiliary Building, 1 st level hallway | 1,2,3,4,5 |
| Reactor Auxiliary Building, 2 nd level hallway | 1,2,3,4,5 |
| Charging Pump Room | 1,2,3,4,5 |
| Component Cooling Water Pump Room | 1,2,3,4,5 |
| Primary Sample Room | 1,2,3,4,5 |
| Primary Demineralizer Room | 1,2,3 |
| Spent Fuel Pump / Heat Exchanger Room | 1,2,3,4,5 |
| Pipe Alley | 4 |
| RHR Heat Exchanger Room | 4 |
| RHR Pump Room entry area (access to RHR Pump CCW flow indication / control) | 4 |
| Boric Acid Batch Tank Room | 1,2,3,4,5 |
| Emergency Bus E1/E2 Room | 3,4,5 |
| Turbine Building 1 st Floor (includes Condensate Polisher, Makeup Water Treatment and Secondary Sample Room) | 1,2,3,4 |
| Turbine Building 2 nd Floor | 1,2,3,4 |
| Turbine Building 3 rd Floor | 1,3,4 |
| Containment Building | 3 |

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EAL Bases

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).

RNP Basis:

If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

The list of plant rooms or areas with entry-related mode applicability identified specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

A list of hazardous gases (ref. 2, 3):

| Flammable Gas (1) or Toxic (2) | Asphyxiant Gas |
|---------------------------------------|--|
| Acetylene (1, 2) | Nitrogen |
| Oxygen (1) | Argon |
| Propane (1, 2) | Carbon Dioxide |
| Hydrogen (1) | Halon |
| Ammonia (1, 2) | Helium Freon – Genetron Dichlorodifluormethane |
| P-10 Gas, used in portal monitors (1) | Freon – R-22, Chlorodifluoromethane |
| Ethanolamine (1, 2) | |
| Methoxypropylamine (1, 2) | |
| Dimethylamine (1, 2) | |

NEI 99-01 Basis:

This IC addresses an event involving a release of a hazardous gas that precludes or impedes access to equipment necessary to maintain normal plant operation, or required for a normal plant cooldown and shutdown. This condition represents an actual or potential substantial degradation of the level of safety of the plant.

An Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous

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EAL Bases

release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.

Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the ~~Emergency Director~~ **Emergency Coordinator's** judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

An emergency declaration is not warranted if any of the following conditions apply:

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.
- **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 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 **AR**, C or F ICs.

RNP Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases
2. PLP-021, Chemical Storage, Inventory, Spill and Hazard Communication Program

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3. PLP-022, Environmental Regulatory Compliance Guidelines for Disposal of Hazardous Waste/Surplus Chemicals
4. NEI 99-01 HA5

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EAL Bases

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 Dedicated/Alternate Shutdown System

Mode Applicability:

All

Definition(s):

None

RNP Basis:

The Shift Manager (SM) determines if the Control Room is uninhabitable and requires evacuation. Control Room uninhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (ref. 1, 2).

Inability to establish plant control from outside the Control Room escalates this event to a Site Area Emergency per EAL HS6.1.

NEI 99-01 Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Escalation of the emergency classification level would be via IC HS6.

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EAL Bases

RNP Basis Reference(s):

1. AOP-004, Control Room Inaccessibility
2. DSP-002, Hot Shutdown Using the Dedicated/Alternate Shutdown System
3. NEI 99-01 HA6

ATTACHMENT 1

EAL Bases

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 Dedicated/Alternate Shutdown System

AND

Control of **any** of the following key safety functions is not reestablished within 15 min.
(Note 1):

- Reactivity
- Core Cooling
- RCS heat removal

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

All

Definition(s):

None

RNP Basis:

The Shift Manager determines if the Control Room is uninhabitable and requires evacuation. Control Room inhabitable may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (ref. 1, 2).

NEI 99-01 Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not “control” is established at the remote safe shutdown location(s) is based on Emergency ~~Director-Coordinator~~ judgment. The Emergency ~~Director-Coordinator~~ is expected to make a reasonable, informed judgment within ~~(the site-specific time for transfer)~~ 15 minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

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EAL Bases

Escalation of the emergency classification level would be via IC FG1 or CG1

RNP Basis Reference(s):

1. AOP-004, Control Room Inaccessibility
2. DSP-002, Hot Shutdown Using the Dedicated/Alternate Shutdown System
3. NEI 99-01 HS6

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EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a UE

EAL:

HU7.1 Unusual Event

Other conditions exist which in the judgment of the Emergency Coordinator 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.

RNP Basis:

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the RNP Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage

ATTACHMENT 1

EAL Bases

the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

NEI 99-01 Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency ~~Director~~ ~~Coordinator~~ to fall under the emergency classification level description for an ~~NOUE~~ Unusual Event.

RNP Basis Reference(s):

1. PLP-007, Robinson Emergency Plan
2. NEI 99-01 HU7

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EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions exist that in the judgment of the Emergency Coordinator warrant declaration of an Alert

EAL:

HA7.1 Alert

Other conditions exist which, in the judgment of the Emergency Coordinator, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. **Any** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward RNP 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 RNP. 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).

RNP Basis:

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the RNP Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref.1).

NEI 99-01 Basis:

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EAL Bases

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~~-Coordinator to fall under the emergency classification level description for an Alert.

RNP Basis Reference(s):

1. PLP-007, Robinson Emergency Plan
2. NEI 99-01 HA7

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EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a Site Area Emergency

EAL:

HS7.1 Site Area Emergency

Other conditions exist which in the judgment of the Emergency Coordinator 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 RNP 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 RNP. 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).

SITE BOUNDARY - As depicted in UFSAR Figure 2.1.1-4, Plant Site Boundary and Exclusion Zone. For the purpose of dose assessment the 'site boundary' is considered to be a 0.265 mile (1400 ft.) radius around the plant.

RNP Basis:

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the RNP Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage

ATTACHMENT 1
EAL Bases

the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

NEI 99-01 Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency ~~Director~~-Coordinator to fall under the emergency classification level description for a Site Area Emergency.

RNP Basis Reference(s):

1. PLP-007, Robinson Emergency Plan
2. NEI 99-01 HS7

ATTACHMENT 1

EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Coordinator warrant declaration of a General Emergency

EAL:

HG7.1 General Emergency

Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward RNP 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 RNP. 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.

RNP Basis:

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the RNP Emergency Response Plan. The Operations Shift Manager(SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

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Releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the SITE BOUNDARY.

NEI 99-01 Basis:

This IC addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency ~~Director~~ Coordinator to fall under the emergency classification level description for a General Emergency.

RNP Basis Reference(s):

1. PLP-007, Robinson Emergency Plan
2. NEI 99-01 HG7

ATTACHMENT 1 EAL Bases

Category S – System Malfunction

EAL Group: Hot Conditions (RCS temperature > 200°F); EALs in this category are applicable only in one or more hot operating modes.

Numerous system-related equipment failure events that warrant emergency classification have been identified in this category. They may pose actual or potential threats to plant safety.

The events of this category pertain to the following subcategories:

1. Loss of Emergency AC Power

Loss of emergency electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite sources for 480V emergency buses.

2. Loss of Vital DC Power

Loss of emergency electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of vital plant 125 VDC power sources.

3. Loss of Control Room Indications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant increase from these base-line levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

5. RCS Leakage

The reactor vessel provides a volume for the coolant that covers the reactor core. The reactor 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

ATTACHMENT 1

EAL Bases

cracks that may propagate to an extent threatening fuel clad, RCS and containment integrity.

6. RPS Failure

This subcategory includes events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RPS to complete a reactor trip comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean any trip failure event that does not achieve reactor shutdown (< 5% reactor power). If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and containment integrity.

7. Loss of Communications

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

8. Containment Failure

Failure of containment isolation capability (under conditions in which the containment is not currently challenged) or loss of containment depressurization system capability warrants emergency classification.

9. Hazardous Event Affecting Safety Systems

Various natural and technological events that result in degraded plant safety system performance or significant visible damage warrant emergency classification under this subcategory.

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EAL Bases

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite AC power capability to emergency buses for 15 minutes or longer

EAL:

SU1.1 Unusual Event

Loss of **all** offsite AC power capability to 480V emergency buses E-1 and E-2 for ≥ 15 min. (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

RNP Basis:

Emergency buses E-1 and E-2 are the essential buses.

For emergency classification purposes, “capability” means that an AC power source is available to the emergency buses, whether or not the buses are powered from it.

Offsite power is connected to the 230KV switchyard through six lines:

- Darlington County Plant south through 230 KV CB 52/11 and 230 KV CB 52/12
- Darlington County Plant north through 230 KV CB 52/13 and 230 KV CB 52/14
- Sumter through 230 KV CB 52/10 and 230 KV CB 52/11
- Rockingham through 230 KV CB 52/2 and 230 KV CB 52/3
- Darlington SCPSA through 230 KV CB 52/7 and 230 KV CB 52/8
- Florence through 230 KV CB 52/4 and 230 KV CB 52/5

The Unit 1 115KV switchyard is connected to the 230KV through two lines:

- #1 Auto transformer to 115 KV west bus through 230 KV CBs 52/5 and 52/6
- #2 Auto transformer to 115 KV east bus through 230 KV CBs 52/1 and 52/2

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Emergency bus E-1 is supplied from 4160V bus 2 through Station Service Transformer 2F (SST 2F). When the Main Generator is in service, 4160V bus 2 is supplied from the Unit Auxiliary Transformer through 4160V bus 1 and tie breaker 52/10. When the Main Generator is shutdown, 4160V bus 2 is supplied from the Startup Transformer. Emergency bus E-2 is supplied from 4160V bus 3 through Station Service Transformer 2G (SST 2G). 4160V bus 3 is normally supplied from the Startup Transformer. The Startup Transformer is supplied from the lines connecting to the 115KV grid and two ties to the 230KV grid.

Another method to obtain offsite power is by backfeeding the emergency buses through the Main and Unit Auxiliary Transformer. This is only done during Cold Shutdown unless nuclear safety considerations require it to be done during Hot Shutdown when no other power sources are available. Buses E-1 and E-2 can also be supplied from onsite Emergency Diesel Generators (EDGs) A and B, respectively (ref.1, 2, 3, 4, 5, 6).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses.

NEI 99-01 Basis:

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC emergency buses. This condition represents a potential reduction in the level of safety of the plant.

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the emergency buses, whether or not the buses are powered from it.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of offsite power.

Escalation of the emergency classification level would be via IC SA1.

RNP Basis Reference(s):

1. OP-603, Electrical Distribution
2. UFSAR Figure 8.1.2-1a
3. UFSAR Section 8.3 Onsite Power Systems
4. AOP-024, Loss of Instrument Bus
5. DBD/R87038/SD16, Electrical Distribution System
6. EOP-ECA-0.0, Loss of All AC Power
7. NEI 99-01 SU1

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all but one** AC power source to emergency buses for 15 minutes or longer

EAL:

SA1.1 Alert

AC power capability to 480V emergency buses E-1 and E-2 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 Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

- (1) The integrity of the reactor coolant pressure boundary;
- (2) The capability to shut down the reactor and maintain it in a safe shutdown condition;
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

RNP Basis:

Emergency buses E-1 and E-2 are the essential buses.

For emergency classification purposes, “capability” means that an AC power source is available to the emergency buses, whether or not the buses are powered from it.

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Offsite power is connected to the 230KV switchyard through six lines:

- Darlington County Plant south through 230 KV CB 52/11 and 230 KV CB 52/12
- Darlington County Plant north through 230 KV CB 52/13 and 230 KV CB 52/14
- Sumter through 230 KV CB 52/10 and 230 KV CB 52/11
- Rockingham through 230 KV CB 52/2 and 230 KV CB 52/3
- Darlington SCPSA through 230 KV CB 52/7 and 230 KV CB 52/8
- Florence through 230 KV CB 52/4 and 230 KV CB 52/5

The Unit 1 115KV switchyard is connected to the 230KV through two lines:

- #1 Auto transformer to 115 KV west bus through 230 KV CBs 52/5 and 52/6
- #2 Auto transformer to 115 KV east bus through 230 KV CBs 52/1 and 52/2

Emergency bus E-1 is supplied from 4160V bus 2 through Station Service Transformer 2F (SST 2F). When the Main Generator is in service, 4160V bus 2 is supplied from the Unit Auxiliary Transformer through 4160V bus 1 and tie breaker 52/10. When the Main Generator is shutdown, 4160V bus 2 is supplied from the Startup Transformer. Emergency bus E-2 is supplied from 4160V bus 3 through Station Service Transformer 2G (SST 2G). 4160V bus 3 is normally supplied from the Startup Transformer. The Startup Transformer is supplied from the lines connecting to the 115KV grid and two ties to the 230KV grid.

Another method to obtain offsite power is by backfeeding the emergency buses through the Main and Unit Auxiliary Transformer. This is only done during Cold Shutdown unless nuclear safety considerations require it to be done during Hot Shutdown when no other power sources are available. Buses E-1 and E-2 can also be supplied from onsite Emergency Diesel Generators (EDGs) A and B, respectively (ref.1, 2, 3, 4, 5, 6).

NEI 99-01 Basis:

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

An “AC power source” is a source recognized in AOPs and EOPs, and capable of supplying required power to an emergency bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from the unit main generator.

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- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from an offsite power source.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

Escalation of the emergency classification level would be via IC SS1.

RNP Basis Reference(s):

1. OP-603, Electrical Distribution
2. UFSAR Figure 8.1.2-1a
3. UFSAR Section 8.3 Onsite Power Systems
4. AOP-024, Loss of Instrument Bus
5. DBD/R87038/SD16, Electrical Distribution System
6. EOP-ECA-0.0, Loss of All AC Power
7. NEI 99-01 SA1

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Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite power and **all** onsite AC power to emergency buses for 15 minutes or longer

EAL:

SS1.1 Site Area Emergency

Loss of **all** offsite and **all** onsite AC power capability to 480V emergency buses E-1 and E-2 for ≥ 15 min. (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

Emergency buses E-1 and E-2 are the essential buses.

This EAL is indicated by the loss of all offsite and onsite AC power capability to 480V emergency buses E-1 and E-2. For emergency classification purposes, “capability” means that an AC power source is available to the emergency buses, whether or not the buses are powered from it.

Offsite power is connected to the 230KV switchyard through six lines:

- Darlington County Plant south through 230 KV CB 52/11 and 230 KV CB 52/12
- Darlington County Plant north through 230 KV CB 52/13 and 230 KV CB 52/14
- Sumter through 230 KV CB 52/10 and 230 KV CB 52/11
- Rockingham through 230 KV CB 52/2 and 230 KV CB 52/3
- Darlington SCPSA through 230 KV CB 52/7 and 230 KV CB 52/8
- Florence through 230 KV CB 52/4 and 230 KV CB 52/5

The Unit 1 115KV switchyard is connected to the 230KV through two lines:

- #1 Auto transformer to 115 KV west bus through 230 KV CBs 52/5 and 52/6

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- #2 Auto transformer to 115 KV east bus through 230 KV CBs 52/1 and 52/2

Emergency bus E-1 is supplied from 4160V bus 2 through Station Service Transformer 2F (SST 2F). When the Main Generator is in service, 4160V bus 2 is supplied from the Unit Auxiliary Transformer through 4160V bus 1 and tie breaker 52/10. When the Main Generator is shutdown, 4160V bus 2 is supplied from the Startup Transformer. Emergency bus E-2 is supplied from 4160V bus 3 through Station Service Transformer 2G (SST 2G). 4160V bus 3 is normally supplied from the Startup Transformer. The Startup Transformer is supplied from the lines connecting to the 115KV grid and two ties to the 230KV grid.

Another method to obtain offsite power is by backfeeding the emergency buses through the Main and Unit Auxiliary Transformer. This is only done during Cold Shutdown unless nuclear safety considerations require it to be done during Hot Shutdown when no other power sources are available. Buses E-1 and E-2 can also be supplied from onsite Emergency Diesel Generators (EDGs) A and B, respectively (ref.1, 2, 3, 4, 5, 6).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. The interval begins when both offsite and onsite AC power capability are lost.

NEI 99-01 Basis:

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via ICs **AG4RG1**, FG1 or SG1.

RNP Basis Reference(s):

1. OP-603, Electrical Distribution
2. UFSAR Figure 8.1.2-1a
3. UFSAR Section 8.3 Onsite Power Systems
4. AOP-024, Loss of Instrument Bus
5. DBD/R87038/SD16, Electrical Distribution System
6. EOP-ECA-0.0, Loss of All AC Power
7. NEI 99-01 SS1

ATTACHMENT 1
EAL Bases

Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power capability to 480V emergency buses E-1 and E-2

AND EITHER:

- Restoration of at least one emergency bus in < 8 hours is **not** likely (Note 1)
- Core Cooling **RED** Path entry conditions met

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

This EAL is indicated by the extended loss of all offsite and onsite AC power capability 480V emergency buses E-1 and E-2 either for greater than the RNP Station Blackout (SBO) coping analysis time (8 hrs.) (ref. 7) or that has resulted in indications of an actual loss of adequate core cooling.

Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED Path conditions being met. (ref. 8).

Offsite power is connected to the 230KV switchyard through six lines:

- Darlington County Plant south through 230 KV CB 52/11 and 230 KV CB 52/12
- Darlington County Plant north through 230 KV CB 52/13 and 230 KV CB 52/14
- Sumter through 230 KV CB 52/10 and 230 KV CB 52/11
- Rockingham through 230 KV CB 52/2 and 230 KV CB 52/3
- Darlington SCPSA through 230 KV CB 52/7 and 230 KV CB 52/8

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- Florence through 230 KV CB 52/4 and 230 KV CB 52/5

The Unit 1 115KV switchyard is connected to the 230KV through two lines:

- #1 Auto transformer to 115 KV west bus through 230 KV CBs 52/5 and 52/6
- #2 Auto transformer to 115 KV east bus through 230 KV CBs 52/1 and 52/2

Emergency bus E-1 is supplied from 4160V bus 2 through Station Service Transformer 2F (SST 2F). When the Main Generator is in service, 4160V bus 2 is supplied from the Unit Auxiliary Transformer through 4160V bus 1 and tie breaker 52/10. When the Main Generator is shutdown, 4160V bus 2 is supplied from the Startup Transformer. Emergency bus E-2 is supplied from 4160V bus 3 through Station Service Transformer 2G (SST 2G). 4160V bus 3 is normally supplied from the Startup Transformer. The Startup Transformer is supplied from the lines connecting to the 115KV grid and two ties to the 230KV grid.

Another method to obtain offsite power is by backfeeding the emergency buses through the Main and Unit Auxiliary Transformer. This is only done during Cold Shutdown unless nuclear safety considerations require it to be done during Hot Shutdown when no other power sources are available. Buses E-1 and E-2 can also be supplied from onsite Emergency Diesel Generators (EDGs) A and B, respectively (ref. 1, 2, 3, 4, 5, 6).

Eight hours is the station blackout coping time (ref 7).

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on Emergency Coordinator judgment as it relates to imminent Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers. Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED Path conditions being met (ref. 8). Specifically, Core Cooling RED Path conditions exist if either core exit T/Cs are reading greater than or equal to 1200°F or subcooling is less than 18°F [37°F] AND no RCPs are running AND core exit T/Cs are reading greater than or equal to 700°F AND RVLIS Full Range is less than 41% (ref. 8).

NEI 99-01 Basis:

This IC addresses a prolonged loss of all power sources to AC emergency buses. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A prolonged loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

The EAL should require declaration of a General Emergency prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC emergency bus by the end of the analyzed station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of challenges to multiple fission product barriers.

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EAL Bases

The estimate for restoring at least one emergency bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, protective actions for the public.

The EAL will also require a General Emergency declaration if the loss of AC power results in parameters that indicate an inability to adequately remove decay heat from the core.

RNP Basis Reference(s):

1. OP-603, Electrical Distribution
2. UFSAR Figure 8.1.2-1a
3. UFSAR Section 8.3 Onsite Power Systems
4. AOP-024, Loss of Instrument Bus
5. DBD/R87038/SD16, Electrical Distribution System
6. EOP-ECA-0.0, Loss of All AC Power
7. 8S19-P-101, H.B. Robinson, Unit No. 2 Station Blackout Coping Analysis Report
8. Critical Safety Function Status Trees, CSF-2 Core Cooling
9. NEI 99-01 SG1

ATTACHMENT 1
EAL Bases

Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** 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 480V emergency buses E-1 and E-2 for ≥ 15 min.

AND

Loss of **all** vital DC power based on < 109.5 VDC Bus A and < 106.2 VDC Bus B voltage indications for ≥ 15 min.

(Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

This EAL is indicated by the loss of all offsite and onsite emergency AC power capability to 480V emergency buses E-1 and E-2 for greater than 15 minutes in combination with degraded vital DC power voltage. This EAL addresses operating experience from the March 2011 accident at Fukushima Daiichi.

Offsite power is connected to the 230KV switchyard through six lines:

- Darlington County Plant south through 230 KV CB 52/11 and 230 KV CB 52/12
- Darlington County Plant north through 230 KV CB 52/13 and 230 KV CB 52/14
- Sumter through 230 KV CB 52/10 and 230 KV CB 52/11
- Rockingham through 230 KV CB 52/2 and 230 KV CB 52/3
- Darlington SCPSA through 230 KV CB 52/7 and 230 KV CB 52/8
- Florence through 230 KV CB 52/4 and 230 KV CB 52/5

The Unit 1 115KV switchyard is connected to the 230KV through two lines:

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- #1 Auto transformer to 115 KV west bus through 230 KV CBs 52/5 and 52/6
- #2 Auto transformer to 115 KV east bus through 230 KV CBs 52/1 and 52/2

Emergency bus E-1 is supplied from 4160V bus 2 through Station Service Transformer 2F (SST 2F). When the Main Generator is in service, 4160V bus 2 is supplied from the Unit Auxiliary Transformer through 4160V bus 1 and tie breaker 52/10. When the Main Generator is shutdown, 4160V bus 2 is supplied from the Startup Transformer. Emergency bus E-2 is supplied from 4160V bus 3 through Station Service Transformer 2G (SST 2G). 4160V bus 3 is normally supplied from the Startup Transformer. The Startup Transformer is supplied from the lines connecting to the 115KV grid and two ties to the 230KV grid.

Another method to obtain offsite power is by backfeeding the emergency buses through the Main and Unit Auxiliary Transformer. This is only done during Cold Shutdown unless nuclear safety considerations require it to be done during Hot Shutdown when no other power sources are available. Buses E-1 and E-2 can also be supplied from onsite Emergency Diesel Generators (EDGs) A and B, respectively (ref.1, 2, 3, 4, 5, 6).

The A and B batteries are safety-related and are equipped with two redundant battery chargers per bus. The A and B batteries are sized to carry expected shutdown loads following a design basis accident with no battery chargers available for a period of 1 hour without battery terminal voltage falling below minimum allowable voltage. The four safety-related chargers are sized to charge a partially discharged battery within 24 hours while carrying its normal load.

Minimum battery terminal voltage is 1.75 VDC per cell for each of 60 cells per battery or 105 VDC (ref. 3). Calculations performed for the B battery replacement, however, specify minimum battery terminal voltage of 106.8 VDC and a corresponding bus voltage of 106.2 VDC (ref. 7). Battery A minimum bus voltage was calculated to be 109.5 VDC (ref. 8). Control Room annunciator APP-036-D3, BATT A/B LO VOLT, is received at 123 VDC and signals sustained loss of a battery charger or battery/cell failure (ref. 9). Battery bus voltage is indicated on ERFIS Points APV3022A (MCC-A) and APV3023A (MCC-B).

NEI-9901 Basis:

This IC addresses a concurrent and prolonged loss of both emergency AC and Vital DC power. A loss of all emergency AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both emergency AC and **vital** DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

ATTACHMENT 1
EAL Bases

RNP Basis Reference(s):

1. OP-603, Electrical Distribution
2. UFSAR Figure 8.1.2-1a
3. UFSAR Section 8.3 Onsite Power Systems
4. AOP-024, Loss of Instrument Bus
5. DBD/R87038/SD16, Electrical Distribution System
6. EOP-ECA-0.0, Loss of All AC Power
7. Calculation RNP-E-6.031, Station Battery B Replacement, Figure 1
8. Calculation RNP-E-6.018, Section 5.1.4
9. APP-036-D3, BATT A/B LO VOLT
10. NEI 99-01 SG8

ATTACHMENT 1

EAL Bases

Category: S – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

Initiating Condition: Loss of all vital DC power for 15 minutes or longer

EAL:

SS2.1 Site Area Emergency

Loss of **all** vital DC power based on < 109.5 VDC Bus A and < 106.2 VDC Bus B voltage indications for ≥ 15 min. (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

The A and B batteries are safety-related and are equipped with two redundant battery chargers per bus. The A and B batteries are sized to carry expected shutdown loads following a design basis accident with no battery chargers available for a period of 1 hour without battery terminal voltage falling below minimum allowable voltage. The four safety-related chargers are sized to charge a partially discharged battery within 24 hours while carrying its normal load.

Minimum battery terminal voltage is 1.75 VDC per cell for each of 60 cells per battery or 105 VDC (ref. 1). Calculations performed for the B battery replacement, however, specify minimum battery terminal voltage of 106.8 VDC and a corresponding bus voltage of 106.2 VDC (ref. 2). Battery A minimum bus voltage was calculated to be 109.5 VDC (ref. 3). Control Room annunciator APP-036-D3, BATT A/B LO VOLT, is received at 123 VDC and signals sustained loss of a battery charger or battery/cell failure (ref. 4). Battery bus voltage is indicated on ERFIS Points APV3022A (MCC-A) and APV3023A (MCC-B).

ATTACHMENT 1
EAL Bases

NEI 99-01 Basis:

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 ~~AG1RG1~~, FG1 or ~~SG8SG1~~.

RNP Basis Reference(s):

1. UFSAR Section 8.3 Onsite Power Systems
2. Calculation RNP-E-6.031, Station Battery B Replacement, Figure 1
3. Calculation RNP-E-6.018, Section 5.1.4
4. APP-036-D3, BATT A/B LO VOLT
5. NEI 99-01 SS8

ATTACHMENT 1
EAL Bases

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-1 parameters from within the Control Room for ≥ 15 min. (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 Safety System Parameters

- Reactor power
- RCS level
- RCS pressure
- Core exit T/C temperature
- Level in at least one SG
- Auxiliary feed flow in at least one SG

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

RNP Basis:

SAFETY SYSTEM parameters listed in Table S-1 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. SPDS/ERFIS plant computer serve as a redundant compensatory indicators which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

ATTACHMENT 1

EAL Bases

NEI 99-01 Basis:

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an “inability to monitor” means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling ~~[PWR] / RPV level [BWR]~~ and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel 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 ~~SA2SA3~~.

RNP Basis Reference(s):

1. AOP-025, RTGB Instrument Failure
2. AOP-024, Loss of Instrument Bus
3. NEI 99-01 SU2

ATTACHMENT 1
EAL Bases

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-1 parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

Any significant transient is in progress, Table S-2

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 Safety System Parameters

- Reactor power
- RCS level
- RCS pressure
- Core exit T/C temperature
- Level in at least one SG
- Auxiliary feed flow in at least one SG

Table S-2 Significant Transients

- Reactor trip
- Runback > 25% thermal power
- Electrical load rejection > 25% electrical load
- Safety injection actuation

ATTACHMENT 1 EAL Bases

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

RNP Basis:

SAFETY SYSTEM parameters listed in Table S-1 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. SPDS/ERFIS plant computer serve as a redundant compensatory indicators which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

Significant transients are listed in Table S-2 and include response to automatic or manually initiated functions such as reactor trips, runbacks involving greater than 25% thermal power change, electrical load rejections of greater than 25% full electrical load or SI injection actuations.

NEI 99-01 Basis:

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an “inability to monitor” means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling ~~[PWR] / RPV level [BWR]~~ and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to

ATTACHMENT 1

EAL Bases

determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level ~~[PWR] / RPV water level [BWR]~~ cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via ICs FS1 or IC ~~AS1~~RS1

RNP Basis Reference(s):

1. AOP-025, RTGB Instrument Failure
2. AOP-024, Loss of Instrument Bus
3. NEI 99-01 SA2

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

| | |
|--------------|----------------------|
| SU4.1 | Unusual Event |
|--------------|----------------------|

| | |
|--|---|
| | RCS activity > Technical Specification Section 3.4.16 limits (Note 9) |
|--|---|

Note 9: Mode 3 applicable only when RCS temperature is $\geq 500^{\circ}\text{F}$.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby

Definition(s):

None

RNP Basis:

This EAL addresses reactor coolant samples exceeding Technical Specification 3.4.8 which are applicable in Modes 1, 2, 3 and 4. The Technical Specification limits accommodate an iodine spike phenomenon that may occur following changes in thermal power. The Technical Specification LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident (ref. 1).

Consistent with the Technical Specification 3.4.16 RCS activity limit applicability, this EAL is only applicable in Modes 1, 2 and Mode 3 when RCS temperature is $\geq 500^{\circ}\text{F}$ (ref. 1).

NEI 99-01 Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category **A-R** ICs.

RNP Basis Reference(s):

1. Technical Specification 3.4.16 RCS Specific Activity
2. NEI 99-01 SU3

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

| | |
|--------------|----------------------|
| SU4.2 | Unusual Event |
|--------------|----------------------|

| |
|---|
| With letdown in service, letdown line area radiation monitor R-9 > 500 mR/hr (Note 9) |
|---|

Note 9: Mode 3 applicable only when RCS temperature is $\geq 500^{\circ}\text{F}$.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby

Definition(s):

None

RNP Basis:

The normal CVCS charging and letdown flow path allows purification of the reactor coolant and control of the RCS volume. Downstream of the non-regenerative heat exchanger and upstream of the mixed bed demineralizers, the letdown stream passes by area radiation monitor R-9, which is mounted above the letdown line pipe. In order for R-9 readings to represent fission product activity in the reactor coolant and thereby warn of potential fuel element failure, letdown must be in service allowing flow through the letdown line and past the radiation monitor.

The threshold value of 500 mR/hr represents fuel failure in excess of 0.1% and indicates a challenge to the Technical Specifications allowable limits for fuel clad degradation (ref. 1, 2, 3).

Consistent with the Technical Specification 3.4.16 RCS activity limit applicability, this EAL is only applicable in Modes 1, 2 and Mode 3 when RCS temperature is $\geq 500^{\circ}\text{F}$ (ref. 3).

NEI 99-01 Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category **A-R** ICs.

RNP Basis Reference(s):

ATTACHMENT 1
EAL Bases

1. RNP-M/MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels
2. OMM-014, Radiation Monitor Setpoints
3. Technical Specification 3.4.16 RCS Specific Activity
4. NEI 99-01 SU3

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1 Unusual Event

RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min.

OR

RCS identified leakage > 25 gpm for ≥ 15 min.

OR

Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min.

(Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

Water balance inventory calculations are normally used to determine RCS leakage. ERFIS Group Display SP5, RCS Leakage Paths, is used to evaluate parameters that are indicative of an RCS leakage source (ref. 1).

Technical Specifications (ref. 2, 3) defines RCS leakage as follows:

Identified Leakage:

- a. Leakage from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or return) that is captured and conducted to collection systems or a sump or collecting tank, or
- b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage, or
- c. RCS leakage through a steam generator to the Secondary Coolant System (primary-to-secondary leakage).

ATTACHMENT 1 EAL Bases

Unidentified Leakage:

All leakage (except RCP seal water injection or return) that is not identified leakage.

Pressure Boundary Leakage:

Pressure boundary leakage is leakage (except SG leakage) through a nonisolable leak in an RCS component body, pipe wall, or vessel wall.

RCS leakage outside of the containment that is not considered identified or unidentified leakage per Technical Specifications includes leakage via interfacing systems such as RCS to the Component Cooling Water, or systems that directly see RCS pressure outside containment such as Chemical & Volume Control System, Safety Injection, Nuclear Sampling System and Residual Heat Removal System (when in the shutdown cooling mode) (ref. 1, 4).

The existence of leakage from the RCS to the Containment, regardless of the source of leakage, may be detected by one or more of the following conditions (ref. 5):

- The Containment air particulate monitor (R-12) is quite sensitive to low leak rates. The Containment radiogas monitor can be used as a backup to the air particulate monitor.
- A leakage detection system is included which determines leakage losses from water and steam systems within the Containment, including that from the RCS. This system collects and measures moisture condensed from the Containment atmosphere by the cooling coils of the Containment air recirculation cooling units. This system provides a means of measuring leakage, including leaks from the cooling coils themselves which are part of the Containment boundary.
- An increase in the amount of coolant makeup water which is required to maintain normal level in the pressurizer, or an increase in Containment sump level are also used as leakage detection methods.

Escalation of this EAL to the Alert level is via Category F, Fission Product Barrier Degradation, EAL FA1.1.

NEI 99-01 Basis:

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

~~EAL #1 and EAL #2~~The first and second EAL conditions are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications). ~~EAL #3~~The third condition addresses an RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These ~~EALs~~ conditions thus apply to leakage into the containment, a

ATTACHMENT 1

EAL Bases

secondary-side system (e.g., steam generator tube leakage ~~in a PWR~~) or a location outside of containment.

The leak rate values for each ~~EAL-condition~~ were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). ~~EAL #1~~ The first condition uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. ~~For PWRs, a~~ An emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated). ~~For BWRs, a stuck-open Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.~~

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

Escalation of the emergency classification level would be via ICs of Recognition Category ~~A-R~~ or F.

RNP Basis Reference(s):

1. AOP-016, Excessive Primary Plant Leakage
2. Technical Specifications section 1.1 Definitions
3. Technical Specifications 3.4.13, RCS Operational Leakage
4. OST-051, Reactor Coolant System Leakage Evaluation
5. UFSAR Section 5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary
6. NEI 99-01 SU4

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

SU6.1 Unusual Event

An automatic trip did **not** shut down the reactor as indicated by reactor power $\geq 5\%$ after **any** RPS setpoint is exceeded

AND

A subsequent automatic trip or manual trip action taken at the RTGB is successful in shutting down the reactor as indicated by reactor power $< 5\%$ (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

RNP Basis:

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) trip function. A reactor trip is automatically initiated by the RPS when certain continuously monitored parameters exceed predetermined setpoints (ref. 1).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from an automatic reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful trip has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2, 3).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the RTGB (reactor trip pushbuttons). Reactor shutdown achieved by use of other trip actions specified in FRP-S.1 Response to Nuclear Power

ATTACHMENT 1

EAL Bases

Generation/ATWS (remote reactor trip breakers, generator circuit breakers, tripping rod drive motors, tripping the turbine, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 2).

Following any automatic RPS trip signal, EOP-E-0 (ref. 1) and FRP-S.1 (ref. 2) prescribe insertion of redundant manual trip signals to back up the automatic RPS trip function and ensure reactor shutdown is achieved. Even if the first subsequent manual trip signal inserts all control rods to the full-in position immediately after the initial failure of the automatic trip, the lowest level of classification that must be declared is an Unusual Event.

In the event that the operator identifies a reactor trip is imminent and initiates a successful manual reactor trip before the automatic RPS trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. However, if subsequent manual reactor trip actions fail to reduce reactor power below 5%, the event escalates to the Alert under EAL SA6.1.

If by procedure, operator actions include the initiation of an immediate manual trip following receipt of an automatic trip signal and there are no clear indications that the automatic trip failed (such as a time delay following indications that a trip setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic trip or manual actions. If a subsequent review of the trip actuation indications reveals that the automatic trip did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

NEI 99-01 Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor (~~trip [PWR] / scram [BWR]~~) that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic (~~trip [PWR] / scram [BWR]~~) is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor (~~trip [PWR] / scram [BWR]~~), operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor (~~trip [PWR] / scram [BWR]~~)). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor (~~trip [PWR] / scram [BWR]~~) is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor (~~trip [PWR] / scram~~ 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

ATTACHMENT 1

EAL Bases

~~(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 **SA5SA6**. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC **SA5-SA6** or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor ~~(trip [PWR]/scram [BWR])~~ signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor ~~(trip [PWR]/scram [BWR])~~ and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the ~~(trip [PWR]/scram [BWR])~~ failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

RNP Basis Reference(s):

1. EOP-E-0 Reactor Trip or Safety Injection
2. FRP-S.1 Response to Nuclear Power Generation/ATWS
3. CSFST CSF-1 Subcriticality
4. NEI 99-01 SU5

ATTACHMENT 1

EAL Bases

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

SU6.2 Unusual Event

A manual trip did **not** shut down the reactor as indicated by reactor power $\geq 5\%$ after **any** manual trip action was initiated

AND

A subsequent automatic trip or manual trip action taken at the RTGB is successful in shutting down the reactor as indicated by reactor power $< 5\%$ (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

RNP Basis:

This EAL addresses a failure of a manually initiated trip in the absence of having exceeded an automatic RPS trip setpoint and a subsequent automatic or manual trip is successful in shutting down the reactor (reactor power $< 5\%$). (ref. 2).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from a manual reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful trip has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 1, 2, 3).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the RTGB (reactor trip pushbuttons). Reactor shutdown achieved by use of other trip actions specified in FRP-S.1 Response to Nuclear Power Generation/ATWS (remote reactor trip breakers, generator circuit breakers, tripping rod drive

ATTACHMENT 1

EAL Bases

motors, tripping the turbine, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 2).

If both subsequent automatic and subsequent manual reactor trip actions in the Control Room fail to reduce reactor power below the power associated with the safety system design (< 5%) following a failure of an initial manual trip, the event escalates to an Alert under EAL SA6.1

NEI 99-01 Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor (~~trip [PWR] / scram [BWR]~~) that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic (~~trip [PWR] / scram [BWR]~~) is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor (~~trip [PWR] / scram [BWR]~~), operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor (~~trip [PWR] / scram [BWR]~~)). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor (~~trip [PWR] / scram [BWR]~~) is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor (~~trip [PWR] / scram [BWR]~~) using a different switch). Depending upon several factors, the initial or subsequent effort to manually (~~trip [PWR] / scram [BWR]~~) the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor (~~trip [PWR] / scram [BWR]~~) signal. If a subsequent manual or automatic (~~trip [PWR] / scram [BWR]~~) is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor (~~trip [PWR] / scram [BWR]~~)). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

~~Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action. [BWR]~~

The plant response to the failure of an automatic or manual reactor (~~trip [PWR] / scram [BWR]~~) will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA5SA6. Depending upon the plant

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EAL Bases

response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC ~~SA5-SA6~~ or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor (~~trip [PWR]/scram [BWR]~~) signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor (~~trip [PWR]/scram [BWR]~~) and the ~~RPSRTS~~ fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the (~~trip [PWR]/scram [BWR]~~) failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

RNP Basis Reference(s):

1. EOP-E-0 Reactor Trip or Safety Injection
2. FRP-S.1 Response to Nuclear Power Generation/ATWS
3. CSFST CSF-1 Subcriticality
4. NEI 99-01 SU5

ATTACHMENT 1

EAL Bases

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor

EAL:

SA6.1 Alert

An automatic or manual trip fails to shut down the reactor as indicated by reactor power $\geq 5\%$

AND

Manual trip actions taken at the RTGB are **not** successful in shutting down the reactor as indicated by reactor power $\geq 5\%$ (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

RNP Basis:

This EAL addresses any automatic or manual reactor trip signal that fails to shut down the reactor followed by a subsequent manual trip that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed.

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the RTGB (reactor trip pushbuttons). Reactor shutdown achieved by use of other trip actions specified in FRP-S.1 Response to Nuclear Power Generation/ATWS (remote reactor trip breakers, generator circuit breakers, tripping rod drive motors, tripping the turbine, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 2).

5% rated power is a minimum reading on the power range scale that indicates continued power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below 5%, plant response will be similar to that observed during a

ATTACHMENT 1

EAL Bases

normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than 5 % power (ref. 1, 2, 3).

Escalation of this event to a Site Area Emergency would be under EAL SS6.1 or Emergency Coordinator judgment.

NEI 99-01 Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor ~~(trip [PWR] / scram [BWR])~~ that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor ~~(trip [PWR] / scram [BWR])~~). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control 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 shut down the reactor is prolonged enough to cause a challenge to the core cooling ~~[PWR] / RPV water level [BWR]~~ or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS65. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS65 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.

RNP Basis Reference(s):

1. EOP-E-0 Reactor Trip or Safety Injection

| | | |
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|----------|--------|-----------------|

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2. FRP-S.1 Response to Nuclear Power Generation/ATWS
3. CSFST CSF-1 Subcriticality
4. NEI 99-01 SA5

ATTACHMENT 1
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Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual trip fails to shut down the reactor as indicated by reactor power $\geq 5\%$

AND

All actions to shut down the reactor are **not** successful as indicated by reactor power $\geq 5\%$

AND EITHER:

- Core Cooling **RED** Path entry conditions met
- Heat Sink **RED** Path entry conditions met

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

RNP Basis:

This EAL addresses the following:

- Any automatic reactor trip signal followed by a manual trip that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed (EAL SA6.1), **AND**
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS barriers.

Reactor shutdown achieved by use of FRP-S.1 Response to Nuclear Power Generation/ATWS (remote reactor trip breakers, generator circuit breakers, tripping rod drive motors, tripping the turbine, emergency boration or manually driving control rods) are also

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credited as a successful manual trip provided reactor power can be reduced below 5% before indications of an extreme challenge to either core cooling or heat removal exist (ref. 1, 2, 3, 4, 5).

5% rated power is a minimum reading on the power range scale that indicates continued power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below 5%, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than 5% power (ref. 1, 2).

Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED Path conditions being met. Specifically, Core Cooling RED Path conditions exist if either core exit T/Cs are reading greater than or equal to 1200°F or subcooling is less than 18°F [37°F] AND no RCPs are running AND core exit T/Cs are reading greater than or equal to 700°F AND RVLIS Full Range is less than 41% (ref. 4).

Indication of inability to adequately remove heat from the RCS is manifested by CSFST Heat Sink RED Path conditions being met. Specifically, Heat Sink RED Path conditions exist if narrow range level in at least one steam generator is not greater than or equal to 9% [18%] and total feedwater flow to the intact steam generators is less than 300 gpm or 0.2E6 PPH (ref. 5).

NEI 99-01 Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor (~~trip [PWR] / scram [BWR]~~) that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shut down the reactor. The inclusion of this IC and EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Escalation of the emergency classification level would be via IC ~~AG1~~RG1 or FG1.

RNP Basis Reference(s):

1. EOP-E-0 Reactor Trip or Safety Injection
2. FRP-S.1 Response to Nuclear Power Generation/ATWS

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3. CSFST CSF-1 Subcriticality
4. CSFST CSF-2 Core Cooling
5. CSFST CSF-3 Heat Sink
6. NEI 99-01 SS5

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EAL Bases

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

SU7.1 Unusual Event

Loss of **all** Table S-3 onsite communication methods

OR

Loss of **all** Table S-3 offsite communication methods

OR

Loss of **all** Table S-3 NRC communication methods

| Table S-3 Communication Methods | | | |
|--|---------------|----------------|------------|
| System | Onsite | Offsite | NRC |
| Public Address System | X | | |
| PBX Telephone System | X | | |
| Radio Transceivers for RNP and Vicinity | X | | |
| Back-up Telephone System (ESSX) | X | | |
| Plant Security Radio Transceivers | X | | |
| Corporate Telephone Communications System (Voicenet) | | X | X |
| BellSouth | | X | X |
| Dedicated Telephone System to Load Dispatcher | | X | |
| Plant Security Radio Control Station | | X | |
| DEMNET | | X | |
| NRC Emergency Telecommunication System (ETS) | | | X |
| Satellite Phones | | X | X |
| Cellular Phones | | X | X |
| Palmetto 800 Transceivers | | X | |

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

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None

RNP Basis:

Onsite, offsite and NRC communications include one or more of the systems listed in Table S-3 (ref. 1, 2).

The NRC ETS Phone and the NRC HPN Phone are part of the PABX and will be unavailable if the PABX is unavailable.

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

NEI 99-01 Basis:

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to **Offsite Response Organizations (OROs)** and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

EAL #1The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

EAL #2The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are ~~(see Developer Notes)~~ the State, Darlington, Lee and Chesterfield County EOCs.

EAL #3The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

RNP Basis Reference(s):

1. PLP-007, Emergency Plan, Attachment 6.1
2. UFSAR Section 9.5.2 Communications Systems
3. NEI 99-01 SU6

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Category: S – System Malfunction

Subcategory: 8 – Containment Failure

Initiating Condition: Failure to isolate containment or loss of containment pressure control.

EAL:

SU8.1 Unusual Event

EITHER:

Any penetration is **not** isolated within 15 min. of a VALID containment isolation signal

OR

Containment pressure ≥ 10 psig with $<$ one full train of depressurization equipment operating (one CONTAINMENT Spray System train **AND** one CONTAINMENT Cooling System train) per design for ≥ 15 min. (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

VALID - An indication, report, or condition, is considered to be valid when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

RNP Basis:

First Condition:

This threshold addresses incomplete Containment isolation that allows direct release to the environment.

Second Condition:

The Containment Spray System, operating in conjunction with the Containment Cooling System, is designed to cool and depressurize the Containment structure following a Design Basis Accident (ref. 1).

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases requirement. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF

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bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, Containment Spray pump suction is transferred from the RWST to the Containment sump (ref. 2).

The Containment Cooling System consists of two trains of Containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement. Each train of two fan units is supplied with cooling water from a separate train of service water. During normal operation, all four fan units may be operating. In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically if not already running (ref. 2).

The Containment pressure setpoint (10 psig, ref. 3) is the pressure at which the equipment should actuate and begin performing its function. The design basis accident analyses and evaluations assume the loss of one ESF bus resulting in the operation of one Containment Spray System train and one Containment Cooling System train. Consistent with the design requirement, "one full train of depressurization equipment" is therefore defined to be the availability of one train of each system. If less than this equipment is operating and Containment pressure is above the actuation setpoint, the threshold is met.

NEI 99-01 Basis:

This ~~IC~~-EAL addresses a failure of one or more containment penetrations to automatically isolate (close) when required by an actuation signal. It also addresses an event that results in high containment pressure with a concurrent failure of containment pressure control systems. Absent challenges to another fission product barrier, either condition represents potential degradation of the level of safety of the plant.

For ~~EAL #1~~the first condition, the containment isolation signal must be generated as the result on an off-normal/accident condition (e.g., a safety injection or high containment pressure); a failure resulting from testing or maintenance does not warrant classification. The determination of containment and penetration status – isolated or not isolated – should be made in accordance with the appropriate criteria contained in the plant AOPs and EOPs. The 15-minute criterion is included to allow operators time to manually isolate the required penetrations, if possible.

~~EAL #2~~The second condition addresses a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment sprays or ice condenser fans) are either lost or performing in a degraded manner.

This event would escalate to a Site Area Emergency in accordance with IC FS1 if there were a concurrent loss or potential loss of either the Fuel Clad or RCS fission product barriers.

RNP Basis Reference(s):

1. UFSAR Section 6.2.2

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2. Technical Specifications Bases 3.6.6
3. Critical Safety Function Status Tree, CSF-5 Containment
4. NEI 99-01 SU7

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Category: S – System Malfunction

Subcategory: 9 – Hazardous Event Affecting Safety Systems

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

EAL:

SA9.1 Alert

The occurrence of **any** Table S-4 hazardous event

AND EITHER:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused **VISIBLE DAMAGE** to a SAFETY SYSTEM component or structure needed for the current operating mode

| Table S-4 Hazardous Events |
|--|
| <ul style="list-style-type: none">• 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 Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

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FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

- (1) The integrity of the reactor coolant pressure boundary;
- (2) The capability to shut down the reactor and maintain it in a safe shutdown condition;
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

VISIBLE DAMAGE - Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

RNP Basis:

- The significance of seismic events are discussed under EAL HU2.1 (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2).
- The plant Seismic Category I structures are designed to withstand the effects of the design wind, 83 mph (108 gust). (ref. 3, 4).
- An explosion that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

NEI 99-01 Basis:

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

EAL 1.b.1 The first condition addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

EAL 1.b.2 The second condition addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure

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containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC FS1 or ~~AS1~~RS1.

RNP Basis Reference(s):

1. AOP-021 Seismic Disturbances
2. RNP-F/PSA-0009, Assessment of Internal Flooding Events
3. UFSAR Table 3.3.1-1
4. OMM-021, Operation During Adverse Weather Conditions
5. NEI 99-01 CA6

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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 cask/canister must escape its packaging and enter the biosphere for there to be a significant environmental effect resulting from an accident involving the dry storage of spent nuclear fuel. Formal offsite planning is not required because the postulated worst-case accident involving an ISFSI has insignificant consequences to the public health and safety.

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.

Minor surface damage that does not affect storage cask/canister boundary is excluded from the scope of these EALs.

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Category: E - ISFSI

Sub-category: None

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Notification of Unusual Event

Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask > **any** Table E-1 dose limit

| Table E-1 ISFSI Dose Limits | |
|--|---|
| 7P ISFSI | 24P ISFSI |
| <ul style="list-style-type: none">• 400 mrem/hr outside of HSM door on centerline of DSC• 400 mrem/hr at center of air inlets or outlets• 100 mrem/hr on roof, front, back or side | <ul style="list-style-type: none">• 2,600 mrem/hr on the HSM front surface• 10 mrem/hr on the HSM-H door centerline• 20 mrem/hr on the end shield wall exterior |

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 RNP ISFSI, Confinement Boundary is defined as the Dry Shielded Canister (DSC).

RNP Basis:

The ISFSIs (7P and 24P) provide for the dry storage of irradiated fuel assemblies in concrete modules. The principal components are a concrete Horizontal Storage Module (HSM) and a stainless steel Dry Shielded Canister (DSC) with an internal basket which holds the irradiated fuel assemblies. Each HSM contains one DSC and each DSC contains the spent fuel assemblies. The fuel assemblies are confined in a helium atmosphere by the stainless steel canister. Decay heat is removed by thermal radiation, conduction and convection from the canister to an air plenum inside the concrete module. Air flows through this internal plenum by natural draft convection. Both ISFSIs are totally passive system. (ref. 1).

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Confinement boundary is defined as the barrier(s) between areas containing radioactive substances and the environment. Therefore, damage to a confinement boundary must be a confirmed physical breach between the spent fuel and the environment for the DSC.

The values shown in Table E-1 represent 2 times the limits specified in the respective ISFSI (7P and 24P) Technical Specification for radiation external to a loaded HSM for a DSC (ref. 1).

NEI 99-01 Basis:

This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage.

The existence of “damage” is determined by radiological survey. The technical specification multiple of “2 times”, which is also used in Recognition Category ~~A-R~~ IC RAU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the “on-contact” dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

Security-related events for ISFSIs are covered under ICs HU1 and HA1.

RNP Basis Reference(s):

1. H. B. Robinson Independent Spent Fuel Storage Installation SNM-2502 License Appendix A Technical Specification Section 4.2 Limits for the Surface Dose Rate of the HSM during Storage (7P)
2. NGGM-PM-0028 contains CERTIFICATE OF COMPLIANCE NO. 1004 AMENDMENT NO. 10 (24P)
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: The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.
- B. Reactor Coolant System (RCS): The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.
- C. Containment: The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 2). “Loss” and “Potential Loss” signify the relative damage and threat of damage to the barrier. “Loss” means the barrier no longer assures containment of radioactive materials. “Potential Loss” means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Alert:

Any loss or any potential loss of either Fuel Clad or RCS

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

Loss of any two barriers and loss or potential loss of third barrier

The logic used for emergency classification based on fission product barrier monitoring should reflect the following considerations:

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- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier.
- Unusual Event ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.
- For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC RG1 has been exceeded.
- The fission product barrier thresholds specified within a scheme reflect plant-specific RNP 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 Coordinator would have more assurance that there was no immediate need to escalate to a General Emergency.

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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 Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Alert classification level, Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1

NEI 99-01 Basis:

None

RNP Basis Reference(s):

1. NEI 99-01 FA1

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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 Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Site Area Emergency classification level, each barrier is weighted equally. A Site Area Emergency is therefore appropriate for any combination of the following conditions:

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

At the Site Area Emergency classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, the Emergency Coordinator would have greater assurance that escalation to a General Emergency is less imminent.

NEI 99-01 Basis:

None

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RNP Basis Reference(s):

1. NEI 99-01 FS1

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Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss of **any** two barriers and loss or potential loss of third barrier

EAL:

FG1.1 General Emergency

Loss of **any** two barriers

AND

Loss or potential loss of third barrier (Table F-1)

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

RNP Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the General Emergency classification level each barrier is weighted equally. A General Emergency is therefore appropriate for any combination of the following conditions:

- Loss of Fuel Clad, RCS and Containment barriers
- Loss of Fuel Clad and RCS barriers with potential loss of Containment barrier
- Loss of RCS and Containment barriers with potential loss of Fuel Clad barrier
- Loss of Fuel Clad and Containment barriers with potential loss of RCS barrier

NEI 99-01 Basis:

None

RNP Basis Reference(s):

1. NEI 99-01 FG1

ATTACHMENT 2

Fission Product Barrier Loss/Potential Loss Matrix and Bases

Introduction

Table F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Containment). The table is structured so that each of the three barriers occupies adjacent columns. Each fission product barrier column is further divided into two columns; one for Loss thresholds and one for Potential Loss thresholds.

The first column of the table (to the left of the Fuel Clad Loss column) lists the categories (types) of fission product barrier thresholds. The fission product barrier categories are:

- A. RCS or SG Tube Leakage
- B. Inadequate Heat removal
- C. CONTAINMENT Radiation / RCS Activity
- D. CONTAINMENT Integrity or Bypass
- E. Emergency Coordinator Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word “None” is entered in the cell.

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A would be assigned “Fuel Clad Loss A.1,” the third Containment barrier Potential Loss in Category C would be assigned “CONTAINMENT P-Loss C.3,” etc.

If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of Table F-1, locates the likely category and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine if a threshold has been exceeded. If a threshold has not been exceeded, the EAL-user proceeds to the next likely category and continues review of the thresholds in the new category

If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded,

ATTACHMENT 2

Fission Product Barrier Loss/Potential Loss Matrix and Bases

only that one barrier is lost or potentially lost. The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if containment radiation is sufficiently high, a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Containment barrier can occur. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, and FA1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad barrier threshold bases appear first, followed by the RCS barrier and finally the Containment barrier threshold bases. In each barrier, the bases are given according category Loss followed by category Potential Loss beginning with Category A, then B,..., E.

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

| Table F-1 Fission Product Barrier Threshold Matrix | | | | | | |
|---|---|---|---|--|---|--|
| | Fuel Clad Barrier | | Reactor Coolant System Barrier | | Containment Barrier | |
| Category | Loss | Potential Loss | Loss | Potential Loss | Loss | Potential Loss |
| A RCS or SG Tube Leakage | None | None | 1. An automatic or manual ECCS (SI) actuation required by EITHER : <ul style="list-style-type: none"> • UNISOLABLE RCS leakage • SG tube RUPTURE | 1. RCS leakage > capacity of a single charging pump (> 77 gpm) due to EITHER : <ul style="list-style-type: none"> • UNISOLABLE RCS leakage • SG tube leakage 2. CSFST Integrity- RED Path entry conditions met | 1. A leaking or RUPTURED SG is FAULTED outside of containment | None |
| B Inadequate Heat Removal | 1. CSFST Core Cooling- RED Path entry conditions met | 1. CSFST Core Cooling- ORANGE PATH entry conditions met 2. CSFST Heat Sink- RED Path entry conditions met AND Heat sink is required | None | 1. CSFST Heat Sink- RED Path entry conditions met AND Heat sink is required | None | 1. CSFST Core Cooling- RED Path entry conditions met AND Restoration procedures not effective within 15 min. (Note 1) |
| C Cont. Radiation / RCS Activity | 1. Containment High Range Radiation Monitor R-32A or R-32B > 100 R/hr 2. Dose equivalent I-131 coolant activity > 300 µCi/gm | None | 1. Containment High Range Radiation Monitor R-32A or R-32B > 5 R/hr | None | None | 1. Containment High Range Radiation Monitor R-32A or R-32B > 2000 R/hr |
| D Cont. Integrity or Bypass | None | None | None | None | 1. Containment isolation is required AND EITHER : <ul style="list-style-type: none"> • Containment integrity has been lost based on Emergency Coordinator judgment • UNISOLABLE pathway from Containment to the environment exists 2. Indications of RCS leakage outside of Containment | 1. CSFST Containment- RED Path entry conditions met 2. Containment hydrogen concentration ≥ 4% 3. Containment pressure ≥ 10 psig with < one full train of depressurization equipment operating (one CONTAINMENT Spray System train AND one CONTAINMENT Cooling System train) per design for ≥ 15 min. (Note 1) |
| E EC Judgment | 1. Any condition in the opinion of the Emergency Coordinator that indicates loss of the fuel clad barrier | 1. Any condition in the opinion of the Emergency Coordinator that indicates potential loss of the fuel clad barrier | 1. Any condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier | 1. Any condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier | 1. Any condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier | 1. Any condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier |

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: 1. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: 1. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

| |
|---|
| 1. CSFST Core Cooling-RED Path entry conditions met |
|---|

Definition(s):

None

Basis:

Plant-Specific

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path is entered if either (ref. 1):

- Core exit T/Cs are greater than or equal to 1,200°F, or
- Core exit T/Cs are greater than or equal to 700°F with RCS subcooling margin less than or equal to 18°F [37°F], no RCPs are running, and RVLIS full range is less than or equal to 41%.

CSFST values enclosed in brackets apply under adverse containment conditions, which is CV pressure greater than or equal to 4 psig (ref. 2, 3).

Either set of conditions indicates significant core exit superheating and core uncover. This is considered a Loss of the Fuel Clad barrier.

Generic

This reading indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

RNP Basis Reference(s):

1. Critical Safety Function Status Trees (CSFST), CSF-2 Core Cooling
2. OMM-022, EOP Users Guide
3. FRP-C.1, Response to Inadequate Core Cooling
4. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

| |
|--|
| 1. CSFST Core Cooling- ORANGE Path entry conditions met |
|--|

Definition(s):

None

Basis:

Plant-Specific

Critical Safety Function Status Tree (CSFST) Core Cooling-ORANGE path is entered if core exit thermocouples (T/Cs) are less than 1,200°F, RCS subcooling is less than or equal to 18°F [37°F], and any of the following (Ref. 1, 2, 3):

- No RCPs are running and either: core exit T/Cs are greater than or equal to 700°F and RVLIS full range is greater than 41%, or core exit T/Cs are less than 700°F and RVLIS full range is less than or equal to 41%.
- At least one RCP is running and Reactor Vessel water level is less than or equal to RVLIS dynamic head values specified in CSF-2 Core Cooling.

These conditions indicate subcooling has been lost and that some fuel clad damage may potentially occur.

Generic

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

RNP Basis Reference(s):

1. Critical Safety Function Status Trees, CSF-2 Core Cooling
2. FRP-C.2, Response to Degraded Core Cooling
3. OMM-022, EOP User's Guide
4. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. CSFST Heat Sink-RED Path entry conditions met

AND

Heat sink is required

Definition(s):

None

Basis:

Plant-Specific

In combination with RCS Potential Loss B.1, meeting this threshold results in a Site Area Emergency.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED path indicates the ultimate heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

Indication that heat removal is extremely challenged is manifested by entry to CSFST Heat Sink-RED path (Ref. 1, 2, 3). CSFST Heat Sink-RED path is entered if all SGs are less than or equal to 9% [18%] and total FW flow to S/Gs is less than or equal to 300 gpm or 0.2E6 lbm/hr. The combination of these conditions when heat sink is required indicates the heat sink function is under extreme challenge. This condition addresses loss of functions required for Hot Shutdown with the reactor at pressure and temperature and thus is a challenge of the Fuel Clad barrier.

The phrase “and heat sink required” precludes the need for classification for conditions in which either RCS pressure is less than SG pressure or Heat Sink-RED Path entry was created through operator action directed by an EOP. For example, FRP-H.1, Response to Loss of Secondary Heat Sink, specifically states that functional response procedure actions should not be performed if total feed flow capability of 300 gpm is available but total feed flow has been reduced due to operator action as directed by the EOPs. Therefore, Heat Sink Red Path should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification (ref. 2, 3).

Generic

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the Fuel Clad Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

RNP Basis Reference(s):

1. Critical Safety Function Status Trees, CSF-3 Heat Sink
2. FRP-H.1, Response to Loss of Secondary Heat Sink
3. OMM-022, EOP User's Guide
4. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.B

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: C. Containment Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment High Range Radiation Monitor R-32A or R-32B > 100 R/hr

Definition(s):

None

Basis:

Plant-Specific

Containment radiation monitor readings greater than 100 R/hr indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the Containment. From Calculation RNP-M/MECH-1744, "R-32A and R-32B Calculation for Core Damage Assessment," the range of calculated dose rates for 5% fuel gap release for times from 1 hour to 4 hours post reactor trip is 95 R/hr to 900 R/hr. The specified value of 100 R/hr is conservatively at the low end of the calculated range (ref. 1, 2, 3). This value is higher than that specified for RCS barrier Loss #3.

It is important to recognize that the radiation monitor may be sensitive to shine from the Reactor Vessel or RCS piping.

Monitors used for this Fission Product Barrier Loss threshold are the Containment High Range Radiation Monitors R-32A and R-32B. These monitors provide indication in the Control Room with a range of 1E0 to 1E7 R/hr (ref. 4, 5).

Generic

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300 $\mu\text{Ci/gm}$ dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold **3.AC.1** since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the ~~emergency classification level~~ **ECL** to a Site Area Emergency.

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

RNP Basis Reference(s):

1. EPTSC-07, Damage Assessment
2. RNP-M/MECH-1744, R-32A and R-32B Calculation for Core Damage Assessment
3. RNP-M/MECH-1745, Calculation Setpoints for Accident Rad Monitors and EP Declaration levels
4. UFSAR Section 12.3.3.1.2.2
5. OMM-014, Radiation Monitor Setpoints
5. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: C. Containment Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. Dose equivalent I-131 coolant activity > 300 $\mu\text{Ci/gm}$

Definition(s):

None

Basis:

Plant-Specific

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The threshold dose equivalent I-131 concentration is well above that expected for iodine spikes and corresponds to about 5% fuel clad damage. When reactor coolant activity reaches this level the Fuel Clad barrier is considered lost. (ref. 1).

Generic

This threshold indicates that RCS radioactivity concentration is greater than 300 $\mu\text{Ci/gm}$ 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.

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

RNP Basis Reference(s):

1. RNP-M-MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels
2. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.B

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: C. Containment Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: D. Containment Integrity or Bypass

Degradation Threat: Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: D. Containment Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: E. Emergency Coordinator Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Fuel Clad barrier

Definition(s):

None

Basis:

Plant-Specific

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that are to be used by the Emergency **Director Coordinator** in determining whether the Fuel Clad barrier is lost

RNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Fuel Clad Loss 6.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: E. Emergency Coordinator Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Fuel Clad barrier

Basis:

Plant-Specific

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that are to be used by the Emergency ~~Coordinator~~**Director** in determining whether the Fuel Clad barrier is potentially lost. The Emergency ~~Director~~**Coordinator** should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

RNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Potential Fuel Clad Loss 6.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. An automatic or manual ECCS (SI) actuation required by **EITHER:**

- UNISOLABLE RCS leakage
- SG tube RUPTURE

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

RUPTURE - The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

Basis:

Plant-Specific

ECCS (SI) actuation is caused by (ref. 1):

- Pressurizer pressure < 1715 psig
- Containment pressure > 4.0 psig
- Steam Line ΔP > 100 psid
- High steam flow w/ low SG pressure or low RCS T_{avg}

Generic

This threshold is based on an UNISOLABLE RCS leak of sufficient size to require an automatic or manual actuation of the Emergency Core Cooling System (ECCS). This condition clearly represents a loss of the RCS Barrier.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED. If a RUPTURED steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.

RNP Basis Reference(s):

| | | |
|----------|--------|-----------------|
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|----------|--------|-----------------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

1. EOP-E-0 Reactor Trip or Safety Injection
2. EOP-E-3 Steam Generator Tube Rupture
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Loss 1.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

1. RCS leakage > capacity of a single charging pump (> 77 gpm) due to **EITHER:**

- UNISOLABLE RCS leakage
- SG tube leakage

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

Plant-Specific

The Chemical and Volume Control System (CVCS) includes three positive displacement charging pumps each with a capacity of 77 gpm (54 - 69 gpm in the normal charging mode). RCS leakage greater than the capacity of a charging pump is indicative of substantial RCS leakage. (ref. 1, 2).

Generic

This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ECCS (SI) actuation has not occurred. The threshold is met when ~~an operating procedure, or operating crew supervision, directs that a standby charging (makeup) pump be placed in service to restore and maintain pressurizer level~~ it is determined that RCS leakage is greater than the capacity of a single charging pump.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

If a leaking steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.

RNP Basis Reference(s):

1. AOP-16 Excessive Primary Plant Leakage
2. UFSAR Section 9.3.4 Chemical and Volume Control Center

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.A

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

| |
|--|
| 2. CSFST Integrity-RED Path entry conditions met |
|--|

Definition(s):

None

Basis:

Plant-Specific

The "Potential Loss" threshold is defined by the CSFST Reactor Coolant Integrity - RED Path. CSFST RCS Integrity - Red Path plant conditions and associated PTS Limit A indicates an extreme challenge to the safety function when plant parameters are to the left of the limit curve following excessive RCS cooldown (≥ 100 °F/hr) under pressure (ref. 1, 2).

Generic

This condition indicates an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized).

RNP Basis Reference(s):

1. CSFST CSF-4 RCS Integrity
2. CSFST CSF-4a RCS Integrity
2. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.B

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Heat Sink-RED path entry conditions met

AND

Heat sink is required

Definition(s):

None

Basis:

Plant-Specific

In combination with Fuel Clad Potential Loss B.2, meeting this threshold results in a Site Area Emergency.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED path indicates the ultimate heat sink function is under extreme challenge and that the RCS barrier may potentially be lost (ref. 1).

Indication that heat removal is extremely challenged is manifested by entry to CSFST Heat Sink-RED path (Ref. 1, 2, 3). CSFST Heat Sink-RED path is entered if all SGs are less than or equal to 9% [18%] and total FW flow to S/Gs is less than or equal to 300 gpm or 0.2E6 lbm/hr. The combination of these conditions when heat sink is required indicates the heat sink function is under extreme challenge. This condition addresses loss of functions required for Hot Shutdown with the reactor at pressure and temperature and thus is a challenge of the Fuel Clad barrier.

The phrase “and heat sink required” precludes the need for classification for conditions in which either RCS pressure is less than SG pressure or Heat Sink-RED Path entry was created through operator action directed by an EOP. For example, FRP-H.1, Response to Loss of Secondary Heat Sink, specifically states that functional response procedure actions should not be performed if total feed flow capability of 300 gpm is available but total feed flow has been reduced due to operator action as directed by the EOPs. Therefore, Heat Sink Red Path should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification (ref. 2, 3).

Generic

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the RCS Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

Meeting this threshold results in a Site Area Emergency because this threshold is identical to Fuel Clad Barrier Potential Loss threshold ~~2-B.2~~; both will be met. This condition warrants a Site Area Emergency declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase RCS pressure to the point where mass will be lost from the system.

RNP Basis Reference(s):

1. Critical Safety Function Status Trees, CSF-3 Heat Sink
2. FRP-H.1, Response to Loss of Secondary Heat Sink
3. OMM-022, EOP User's Guide
4. NEI 99-01 Inadequate Heat Removal RCS Loss 2.B

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: C. Containment Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

| |
|---|
| 1. Containment High Range Radiation Monitor R-32A or R-32B > 5 R/hr |
|---|

Definition(s):

None

Basis:

Plant-Specific

Containment radiation monitor readings greater than 5 R/hr indicate the release of reactor coolant to the Containment. Due to the normally good fuel conditions and low RCS activity, a significant release of RCS to the Containment may result in dose rates less than the 1 R/hr minimum range of the instrument. Therefore, any positive reading on R-32A or R-32B should be considered a release of RCS to the Containment. Given that the minimum reading of the instrument is 1 R/hr and the instrument range is seven decades, 5 R/hr represents the lowest reading that is considered a clear positive response (ref. 1, 2,3).

The readings are less than those specified for Fuel Clad barrier Loss C.1 because no damage to the fuel clad is assumed. Only leakage from the RCS is assumed for this barrier Loss threshold.

It is important to recognize that the radiation monitor may be sensitive to shine from the Reactor Vessel or RCS piping. Therefore, it is possible that a reading greater than 5 Rem/hr could represent a release from fuel damage into the RCS without a release to Containment.

Monitors used for this Fission Product Barrier Loss threshold are the Containment High Range Radiation Monitors R-32A and R-32B. These monitors provide indication in the Control Room with a range of 1E0 to 1E7 R/hr (Ref. 3). Due to geometry differences, the values calculated for R-32B are approximately 80% of R-32A (ref. 1, 2).

Generic

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold ~~3.AC.1~~ since it indicates a loss of the RCS Barrier only.

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

RNP Basis Reference(s):

1. RNP-M/MECH-1744, R-32A and R-32B Calculation for Core Damage Assessment
2. RNP-M/MECH-1745, Calculation Setpoints for Accident Rad Monitors and EP Declaration levels
3. UFSAR Section 12.3.3.1.2.2
4. OMM-014, Radiation Monitor Setpoints
5. NEI 99-01 CMT Radiation / RCS Activity RCS Loss 3.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: B. Containment Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: D. Containment Integrity or Bypass

Degradation Threat: Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: D. Containment Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: E. Emergency Coordinator Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier

Definition(s):

None

Basis:

Plant-Specific

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the RCS barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to the recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that may be used by the Emergency **Director Coordinator** in determining whether the RCS Barrier is lost.

RNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Loss 6.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: E. Emergency Coordinator Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier

Definition(s):

None

Basis:

Plant-Specific

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the RCS barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to the inability to reach final safety acceptance criteria before completing all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that may be used by the Emergency ~~Director~~ **Coordinator** in determining whether the RCS Barrier is potentially lost. The Emergency ~~Director~~ **Coordinator** should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

RNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Potential Loss 6.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

- | |
|---|
| 1. A leaking or RUPTURED SG is FAULTED outside of containment |
|---|

Definition(s):

FAULTED - The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.

RUPTURED - The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

Basis:

Plant-Specific

None.

Generic

This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for RCS Barrier Potential Loss 4-A.1 and Loss 4-A.1, respectively. This condition represents a bypass of the containment barrier.

FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably (part of the FAULTED definition) and the FAULTED steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition, the steam generator is still considered FAULTED for emergency classification purposes.

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU4 for the fuel

ATTACHMENT 2

Fission Product Barrier Loss/Potential Loss Matrix and Bases

clad barrier (i.e., RCS activity values) and IC SU5 for the RCS barrier (i.e., RCS leak rate values).

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, gland seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category ~~A-R~~ ICs.

The ~~emergency classification level~~ ECLs resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

| P-to-S Leak Rate | Affected SG is FAULTED Outside of Containment? | |
|--|---|---|
| | Yes | No |
| Less than or equal to 25 gpm | No classification | No classification |
| Greater than 25 gpm | Unusual Event per SU4 SU5.1 | Unusual Event per SU4 SU5.1 |
| Requires operation of a standby charging (makeup) pump (<i>RCS Barrier Potential Loss</i>) | Site Area Emergency per FS1.1 | Alert per FA1.1 |
| Requires an automatic or manual ECCS (SI) actuation (<i>RCS Barrier Loss</i>) | Site Area Emergency per FS1.1 | Alert per FA1.1 |

There is no Potential Loss threshold associated with RCS or SG Tube Leakage.

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

RNP Basis Reference(s):

1. EOP-E-2 Faulted Steam Generator Isolation
2. EOP-E-3 Steam Generator Tube Rupture
3. NEI 99-01 RCS or SG Tube Leakage Containment Loss 1.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: B. Inadequate heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Core Cooling-RED Path entry conditions met

AND

Restoration procedures **not** effective within 15 min. (Note 1)

Definition(s):

None

Basis:

Plant-Specific

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path indicates significant core exit superheating and core uncover (ref. 1, 2).

The function restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing (ref. 2).

Generic

This condition represents an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. For this condition to occur, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.

The restoration procedure is considered “effective” if core exit thermocouple readings are decreasing and/or if reactor vessel level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The Emergency ~~Director~~-Coordinator should escalate the emergency classification level as soon as it is determined that the procedure(s) will not be effective.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation in a significant fraction of core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.

RNP Basis Reference(s):

1. CSFST CSF-2 Core Cooling
2. FRP-C.1 Response to Inadequate Core Cooling
3. NEI 99-01 Inadequate Heat Removal Containment Potential Loss 2.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: C. Containment Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

| |
|------|
| None |
|------|

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: C. Containment Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

| |
|--|
| 1. Containment High Range Radiation Monitor R-32A or R-32B > 2000 R/hr |
|--|

Definition(s):

None

Basis:

Plant-Specific

Containment radiation monitor readings greater than 2000 R/hr indicate significant fuel damage, well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier. Per NEI 99-01, the desired value for Containment Potential Loss should correspond to 20% clad damage. The 2000 R/hr threshold is based on taking four times the average calculated values over the various conditions and time frames analyzed in calculation RNP-M/MECH-1744, "R-32A and R-32B Calculation for Core Damage Assessment." (ref. 1, 2).

Even though high radiation levels themselves may not represent a challenge to Containment integrity, the purpose of this criterion is to ensure precautionary public protective actions are taken due to the potential for significant public dose if the activity in the Containment were released. A reading greater than 2000 R/hr on R-31A or R-32B would result in a Loss of clad, Loss of RCS, and a Potential Loss of containment, dictating a General Emergency classification.

It is important to recognize that the radiation monitor may be sensitive to shine from the reactor vessel or RCS piping.

Monitors used for this Fission Product Barrier Loss threshold are Containment High Range Radiation Monitors R-32A and R-32B. These monitors provide indication in the Control Room with a range of 1E0 to 1E7 R/hr (Ref. 3). Due to geometry differences, the calculated values for R-32B are approximately 80% of R-32A (ref. 1, 2, 3).

Generic

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

NUREG-1228, Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the ~~emergency classification level~~ECL to a General Emergency.

RNP Basis Reference(s):

1. RNP-M/MECH-1744, R-32A and R-32B Calculation for Core Damage Assessment
2. RNP-M/MECH-1745, Calculation Setpoints for Accident Rad Monitors and EP Declaration levels
3. UFSAR Section 12.3.3.1.2.2
4. OMM-014, Radiation Monitor Setpoints
5. NEI 99-01 CMT Radiation / RCS Activity Containment Potential Loss 3.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: D. Containment Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. Containment isolation is required

AND EITHER:

- Containment integrity has been lost based on Emergency Coordinator judgment
- UNISOLABLE pathway from containment to the environment exists

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

Plant-Specific

None

Generic

These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both **bulleted** thresholds **4.A.1 and 4.A.2**.

4.A.1 First Threshold – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the Emergency **Director-Coordinator** will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure **9-F-41**. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve.

ATTACHMENT 2

Fission Product Barrier Loss/Potential Loss Matrix and Bases

Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

Another example would be a loss or potential loss of the RCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category **A-R** ICs.

4.A.2Second Threshold – Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment atmosphere to the environment. As used here, the term “environment” includes the atmosphere of a room or area, outside the containment, that may, in turn, communicate with the outside-the-plant atmosphere (e.g., through discharge of a ventilation system or atmospheric leakage). Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.

Refer to the top piping run of Figure **9-F-41**. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Leakage between two interfacing liquid systems, by itself, does not meet this threshold.

Refer to the bottom piping run of Figure **9-F-41**. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump developed a leak that allowed steam/water to enter the Auxiliary Building, then **second threshold-4.B** would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause **the first threshold 4.A.1** to be met as well.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable containment leakage through various penetrations or system components. Minor releases may also occur if a containment isolation valve(s) fails to close but the containment atmosphere escapes to an

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

enclosed system. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category ~~A~~-R ICs.

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold ~~4~~A.1.

RNP Basis Reference(s):

1. NEI 99-01 CMT Integrity or Bypass Containment Loss 4.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: D. Containment Integrity or Bypass

Degradation Threat: Loss

Threshold:

| |
|--|
| 2. Indications of RCS leakage outside of Containment |
|--|

Definition(s):

None

Basis:

Plant-Specific

EOP-ECA-1.2 LOCA Outside Containment (ref. 1) provides instructions to identify and isolate a LOCA outside of the containment. Potential RCS leak pathways outside containment include (ref. 1):

- Residual Heat Removal
- Safety Injection
- Chemical & Volume Control
- RCP seals/seal return
- PZR/RCS sample lines

Generic

Containment sump, temperature, pressure and/or radiation levels will increase if reactor coolant mass is leaking into the containment. If these parameters have not increased, then the reactor coolant mass may be leaking outside of containment (i.e., a containment bypass sequence). Increases in sump, temperature, pressure, flow and/or radiation level readings outside of the containment may indicate that the RCS mass is being lost outside of containment.

Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of RCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside of containment may not increase significantly; however, other unexpected changes in sump levels, area temperatures or pressures, flow rates, etc. should be sufficient to determine if RCS mass is being lost outside of the containment.

Refer to the middle piping run of Figure 9-F-41. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause threshold 4.AD.1 to be met as well.

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

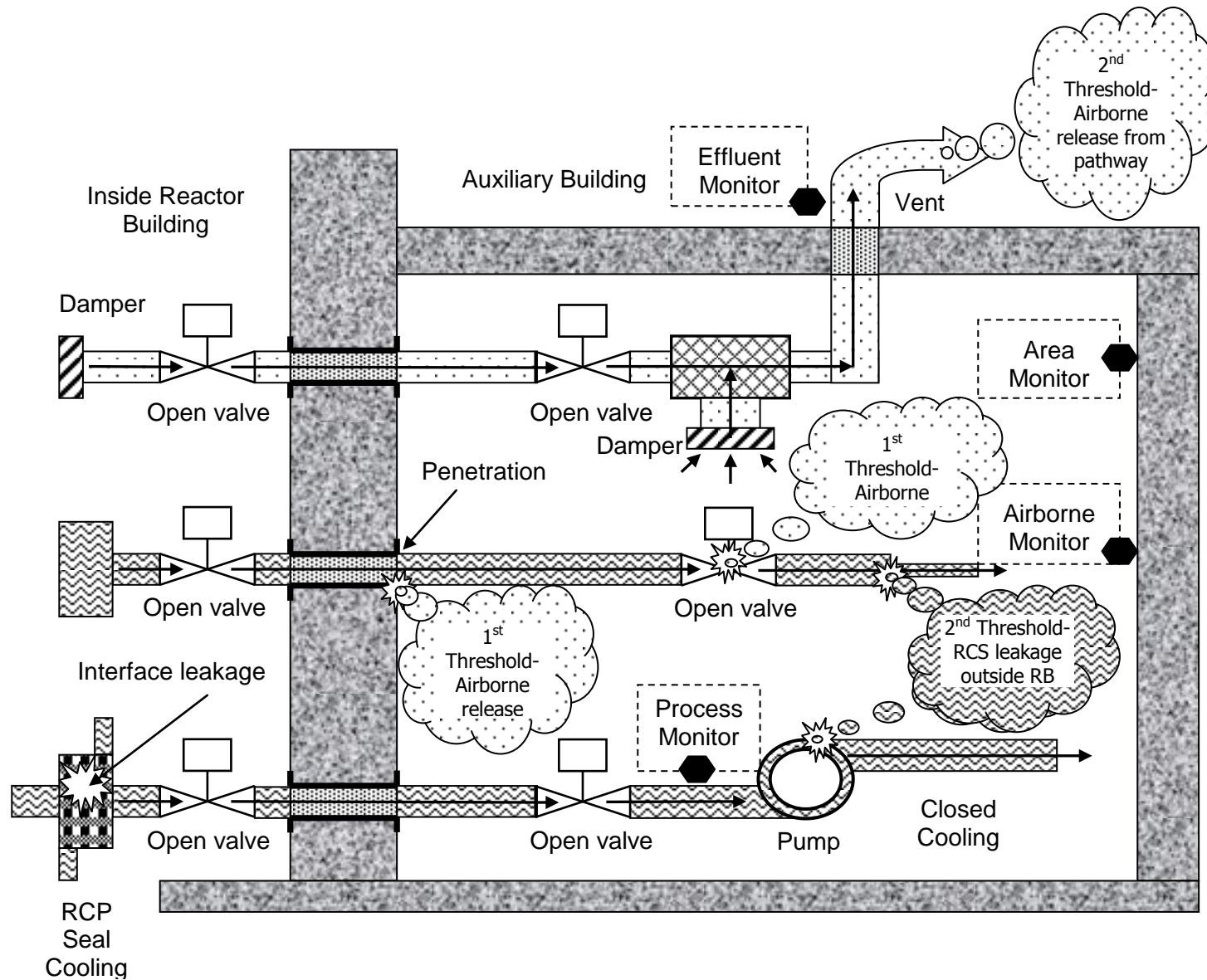
To ensure proper escalation of the emergency classification, the RCS leakage outside of containment must be related to the mass loss that is causing the RCS Loss and/or Potential Loss threshold 4.A.1 to be met.

RNP Basis Reference(s):

1. EOP-ECA-1.2 LOCA Outside Containment
2. NEI 99-01 CMT Integrity or Bypass Containment Loss

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Figure 1: Containment Integrity or Bypass Examples



ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: D. Containment Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

1. CSFST Containment-RED Path entry conditions met

Definition(s):

None

Basis:

Plant-Specific

Critical Safety Function Status Tree (CSFST) Containment-RED Path is entered if containment pressure is greater than or equal to 42 psig and represents an extreme challenge to safety function. (ref. 1, 3).

42 psig is based on the containment design pressure (ref. 3).

Generic

If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the RCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a Site Area Emergency and General Emergency since there is now a potential to lose the third barrier.

RNP Basis Reference(s):

1. CSFST CSF-5 Containment
2. FRP-J.1 Response to High Containment Pressure
3. FSAR 6.2.1.1.5 Acceptance Criteria
4. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: D. Containment Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

| |
|---|
| 2. Containment hydrogen concentration \geq 4% |
|---|

Definition(s):

None

Basis:

Plant-Specific

If hydrogen concentration reaches the lower flammability limit of 4%, (ref. 1) in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside Containment, Loss of the Containment barrier could occur. To generate such levels of combustible gas, Loss of the Fuel Clad and RCS barriers must also have occurred. Since this threshold is also indicative of Loss of both Fuel Clad and RCS barriers with the Potential Loss of the Containment barrier, it therefore will likely warrant declaration of a General Emergency.

Two Containment hydrogen concentration monitors (with a range of 0 to 10% hydrogen) are provided on the Core Cooling and Containment Monitor in the Control Room. Hydrogen concentration is also displayed on ERFIS Points SSC-2512A and SSC-2513A (ref. 2).

Generic

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a potential loss of the Containment Barrier.

RNP Basis Reference(s):

1. CA-3 Hydrogen Flammability in Containment
2. LP-304, Containment Hydrogen Monitor
3. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.B

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment
Category: D. Containment Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

3. Containment pressure ≥ 10 psig with $<$ one full train of depressurization equipment operating (one Containment Spray System train **AND** one Containment Cooling System train) per design for > 15 min. (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Definition(s):

None

Basis:

Plant-Specific

The Containment Spray System, operating in conjunction with the Containment Cooling System, is designed to cool and depressurize the Containment structure following a Design Basis Accident (ref. 1).

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases requirement. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, Containment Spray pump suction is transferred from the RWST to the Containment sump (ref. 2).

The Containment Cooling System consists of two trains of Containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement. Each train of two fan units is supplied with cooling water from a separate train of service water. During normal operation, all four fan units may be operating. In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically if not already running (ref. 2).

The Containment pressure setpoint (10 psig, ref. 3) is the pressure at which the equipment should actuate and begin performing its function. The design basis accident analyses and evaluations assume the loss of one ESF bus resulting in the operation of one Containment Spray System train and one Containment Cooling System train. Consistent with the design requirement, "one full train of depressurization equipment" is therefore defined to be the availability of one train of each system. If less than this equipment is operating and Containment pressure is above the actuation setpoint, the threshold is met.

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Generic

This threshold describes a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems (e.g., containment sprays, ice condenser fans, etc., but not including containment venting strategies) are either lost or performing in a degraded manner.

RNP Basis Reference(s):

1. UFSAR Section 6.2.2
2. Technical Specifications Bases 3.6.6
3. Critical Safety Function Status Tree, CSF-5 Containment
4. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.C

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: F. Emergency Coordinator Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier

Definition(s):

None

Basis:

Plant-Specific

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that may be used by the Emergency **Director Coordinator** in determining whether the Containment Barrier is lost.

RNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: F. Emergency Coordinator Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier

Definition(s):

None

Basis:

Plant-Specific

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that may be used by the Emergency ~~Director~~ **Coordinator** in determining whether the Containment Barrier is lost.

RNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Potential Loss 6.A

ATTACHMENT 3

Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases

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.

ATTACHMENT 3

Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases

RNP Table R-2/H-2 Bases

NEI 99-01 Rev 06 addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant shutdown and cooldown.

The Control Room envelope is automatically placed in the emergency pressurization operating mode as a result of a Safety Injection initiation or Control Room high radiation alarm (R-1). USFSAR chapter 15 analysis demonstrates that dose to Control Room Operators remains within limits during postulated accidents. No special protection against toxic gas intrusion and no toxic gas detectors are provided in the design of the HBR 2 Control Room, however Self-contained breathing apparatuses are available in the Control Room. The buildup of toxic chemical concentrations at the Control Room air intake and within the Control Room volume was evaluated to determine the effect on Control Room habitability from postulated toxic chemical releases. UFSAR Table 6.4.4-3 summarizes the numerical results of this HBR 2 plant toxic chemical habitability analysis and shows compliance with the appropriate limits. Additionally, manual isolation capability via the Emergency Recirculation mode of operation is provided for limiting the intake of hazardous chemicals or smoke. Hazardous chemicals are not stored or transported on or near the site in sufficient quantity as to require isolation capability as a regulatory requirement, however, isolation capability is beneficial and this operational mode is included in the system design to allow the Control Room operators to isolate outside air makeup from the Control Room envelope. Based on these factors the Control Room is excluded from consideration.

Power Operation was reviewed to determine if any actions are “necessary” to maintain power operations. Over reasonable periods there are some actions outside the Control Room that are required to be performed to maintain normal operations. The following table lists the locations into which an operator may be dispatched in order perform a normal plant operation, shutdown or cooldown. The review was completed using the following procedures as the controlling documents:

GP-006-1, Normal Plant Shutdown from Power operation to Hot Shutdown, R9

GP-007, Plant Cooldown from Mode 3 to Mode 5, R101

OP-201, Residual Heat Removal System, R69

Travel paths to the locations where the equipment is operated were considered as part of the determination of affected rooms, RNP Reactor Auxiliary Building design consist of mostly single entry rooms located off of a common hallway, therefore access to the hallway is required to access a given room. Some equipment is located within the hallway itself.

ATTACHMENT 3
Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|---|---|--|--|------|
| GP-006-1 4.6.b | PLACE "B" Mixed Bed (H-OH) in service IAW OP-301-2, Chemical and Volume Control System | | REACTOR AUXILIARY BUILDING | Primary Demineralizer Room | 1 |
| GP-006-1 8.1.4 | NOTIFY E&C to shut down the RCS Zinc Injection System. | | REACTOR AUXILIARY BUILDING | Boric Acid Batch Tank room | 1 |
| GP-006-1 8.1.5 | NOTIFY E&C to control secondary chemical addition IAW CP-SEC-304, Chemical Feed System, during the power reduction. | | 1 st level Turbine Building | Secondary Sample Room & Chemical Feed Room | 1 |
| GP-006-1 8.1.11 | Startup the Auxiliary Boilers. | Associated OP requires alignment of AS system drains. | 1 st level Turbine Building, REACTOR AUXILIARY BUILDING 1 st level hallway | N/A | 1 |
| GP-006-1 8.2.1 | Adjustment of RCP Seal Injection flows (result of raising Letdown flow) | | REACTOR AUXILIARY BUILDING | Charging Pump Room | 1 |
| GP-006-1 8.2.10 | Auxiliary Boiler makeup and modulation | | 1 st level Turbine Building | N/A | 1 |
| GP-006-1 8.2.10 | TRANSFER Gland Sealing Steam to the Auxiliary Steam System IAW of OP-502 | | 1 st level & 2 nd level Turbine Building | N/A | 1 |
| GP-006-1 8.2.10 | Isolation of Extraction Steam | | 2 nd level Turbine Building | N/A | 1 |

ATTACHMENT 3

Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|--|--|--|------|------|
| GP-006-1 8.2.11 | Cylinder heating Steam operation | | 3 rd level Turbine Building | N/A | 1 |
| GP-006-1 8.2.20.c | TRANSFER Steam Generator Blowdown to the Flash Tank with Heat Recovery Bypassed IAW OP-406. | | 1 st level & 2 nd level Turbine Building | N/A | 1 |
| GP-006-1 8.2.20.d | Verify Quenching Valve control switch is in close | | 2 nd level Turbine Building | N/A | 1 |
| GP-006-1 8.2.28.e | IF FCV-1446, CONDENSATE RECIRC, has NOT opened, THEN PERFORM the following to fail open FCV-1446 | | 1 st level Turbine Building | N/A | 1 |
| GP-006-1 8.2.30.a | Locally verify Timer Valves are open | | 3 rd level Turbine Building | N/A | 1 |
| GP-006-1 8.2.35 | Open Unit OCB disconnects | Not required to obtain shutdown or cold shutdown. Action performed in Unit 2 Switchyard, an open area outside of the Protected Area. | Unit 2 Switch Yard | N/A | 2 |
| GP-006-1 8.2.47 | Verify EH Auxiliary Filter in service / air purge | Not required to obtain shutdown or cold shutdown. | 2 nd level Turbine Building | N/A | 3 |
| GP-006-1 8.2.48 | Adjust Cylinder heating Steam | | 3 rd level Turbine Building | N/A | 3 |
| GP-006-1 8.2.49 | Isolate MS-61 / 72 if needed for excess cooldown | | 3 rd level Turbine Building | N/A | 3 |

ATTACHMENT 3

Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|---|--|--|-------------------------------|------|
| GP-006-1 8.2.50.b | Initiate de-gas of RCS | Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Primary sample room, VCT room | 3 |
| GP-006-1 8.2.51 | Isolate SW to Generator H2 coolers | Not required to obtain shutdown or cold shutdown. Only required if Generator H2 pressure will be reduced below SW pressure and leakage exist in H2 coolers | 2 nd & 3 rd level Turbine Building | N/A | 3 |
| GP-006-1 8.2.52 | Isolate SW to secondary coolers | Not required to obtain shutdown or cold shutdown. | 1 st and 2 nd level Turbine Building | N/A | 3 |
| GP-006-1 8.2.53 | Shutdown Turbine Lube Oil | Not required to obtain shutdown or cold shutdown. | 1 st and 2 nd level Turbine Building | N/A | 3 |
| GP-006-1 8.2.54 | Caution Tag Isophase Bus Duct fans | Administrative action, Not required to obtain shutdown or cold shutdown. | 2 nd level Turbine Building | N/A | 3 |
| GP-006-1 8.2.56 | Transfer temperature control to above and below seat drains per OP-405 / adjust SGBD flow | Performed only if removing steam from secondary, not required for normal shutdown & cooldown | 1 st , 2 nd , & 3 rd level Turbine Building | N/A | 3 |
| GP-006-1 8.2.57 | If Condenser vacuum to be broken then and Polishers in service then remove from service | Performed only if removing steam from secondary, not required for normal shutdown & cooldown | 1 st level Turbine Building | Condensate polisher building | 3 |
| GP-006-1 8.2.58 | Shutdown Turbine lube oil, GS, break vacuum | Performed only if removing steam from secondary, not required for normal shutdown & cooldown | 1 st and 2 nd level Turbine Building | | 3 |

ATTACHMENT 3

Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|---|---|--|--|-------|
| GP-006-1 8.3.16 | Isolate De-borating demineralizers | Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Primary Demineralizer Room | 3 |
| GP-006-1 8.3.20.b | Initiate OST-053 | Only initiated for Refueling Shutdown so not required for normal shutdown & cooldown | Containment – all levels | N/A | 3 |
| GP-007 5.11 | Cation or Mixed Bed demineralizer operation | | REACTOR AUXILIARY BUILDING | Primary Demineralizer Room | 3 |
| GP-007 5.13 | Initiate RCS De-gas per OP-918 | Performed prior to solid operation if RCS will be opened. Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Primary Sample Room and VCT Room | 3 |
| GP-007 6.2.2 | Prepare batches of Boric Acid | Maintain Boric Acid inventory | REACTOR AUXILIARY BUILDING | CCW Pump Room and Boric Acid Batch Tank Room | 3/4/5 |
| GP-007 6.2.8 | VERIFY that the Low Temperature Overpressure Protection System is aligned for service when greater than 350°F per OP-006. | Requires verification of pressures and operation of PORV motive force isolation valves in Containment | Containment | | 3 |
| GP-007 6.2.11.d & e | Remove Rod Drive MG sets and RPI from service | Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Rod Drive MG Set Room and 2 nd REACTOR AUXILIARY BUILDING level Hallway | 3 |
| GP-007 6.2.18.b | If SGPORV used for temperature control then station dedicated local operator | | 3 rd level Turbine Building | N/A | 3/4 |

ATTACHMENT 3

Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|---------------------------------------|---|--|--|------|
| GP-007 6.2.22.c | Transfer GS to Auxiliary Steam | | 1 st and 2 nd level Turbine Building | N/A | 3 |
| GP-007 6.2.25.b | Close breakers for SI-865 valves | | REACTOR AUXILIARY BUILDING | 1 st level hallway and E1/E2 Room | 3 |
| GP-007 6.3.1.c/d | Install Steam Dump Jumpers | Not required to obtain shutdown or cold shutdown. Jumper allows operation of 5 dump valves, cooldown can be accomplished with 3 dump valves | E1/E2 Room | | 3 |
| GP-007 6.3.5.b | Open breakers for SI-865 valves | | REACTOR AUXILIARY BUILDING | 1 st level Hallway and E1/E2 room | 3 |
| GP-007 6.3.10.d.1 | Pulling fuses for all but one SI Pump | | REACTOR AUXILIARY BUILDING | E1/E2 Room | 3 |
| GP-007 8.3.10.d.2 | Verification of SI flowpath valves | Valves are being verified in their normal positions, Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Pipe Alley, BIT Room | 3 |
| GP-007 6.3.10.d.3 | Place Caution Tags on SI valves | Administrative action, Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Pipe Alley, BIT Room, E1/E2 Room | 3 |
| GP-007 6.4.1.f | Close breakers for SI-878A/B valves | | REACTOR AUXILIARY BUILDING | 1 st floor hallway, E1/E2 Room | 4 |
| GP-007 6.4.1.i | Place caution tags on SI valves | Administrative action, Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Pipe Alley | 4 |

ATTACHMENT 3

Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|---|---|----------------------------------|--|------|
| GP-007 6.4.1.o/p/ q | Verify SW flow through both CCW HX / Verify both CCW HX in service / Verify 2 CCW Pumps operating | | REACTOR AUXILIARY BUILDING | CCW Pump Room | 4 |
| GP-007 8.4.1.t | Throttle CC-775, as required, to obtain desired flow through RHR HX. | | REACTOR AUXILIARY BUILDING | Spent Fuel Pump / Heat Exchanger Room | 4 |
| GP-007- 6.4.2.b | Align RHR for core cooling (OP-201) | | REACTOR AUXILIARY BUILDING | 1 st level Hallway, RHR pump room deck (access through SFP HX Room), RHR Heat Exchanger Room, Pipe Alley | 4 |
| GP-007 6.4.2.c /d | Isolate standby RHR train | | REACTOR AUXILIARY BUILDING | E1/E2 Room, 1 st level hallway | 4 |
| GP-007 6.4.3.b | Place Caution Tags on selected SI valves | Administrative action only, Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | RHR Pump Room | 4 |
| GP-007 8.4.5 | PIT testing of HCV-758 and FCV-605 | | REACTOR AUXILIARY BUILDING | RHR Heat Exchanger Room | 4 |
| GP-007 6.4.5.e | Close breakers for RHR-750 and RHR- 751 | | REACTOR AUXILIARY BUILDING | 1 st level Hallway, E1/E2 Room | 4 |
| GP-007 6.4.5.i/j/m/n/o | Observe FI-608, Cycle RHR-754A | | REACTOR AUXILIARY BUILDING | Pipe Alley | 4 |
| GP-007 6.4.5.q And 6.4.7.f | Throttle CC-748A/B | | REACTOR AUXILIARY BUILDING | RHR Heat Exchanger Room | 4 |

ATTACHMENT 3

Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|--|--|--|--|-----------|
| GP-007 6.4.7.m | Open CVC-309E | Maximizes letdown flow, Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Non-regenerative Heat exchanger Room | 4 |
| GP-007 6.4.8.a | Locally monitor Feedwater Section Valve Cycling | Administrative action, Not required to obtain shutdown or cold shutdown. | 1 st level Turbine Building | | 4 |
| GP-007 6.4.9 | Open / Throttle MSIV above and below seat drains | | 3 rd level Turbine Building | | 4 |
| GP-007 6.4.9 | Adjust SW header pressure | | REACTOR AUXILIARY BUILDING | CCW Pump Room | 4/5 |
| GP-007 6.4.11.j | Place N2 blanket on SG per OP-406 | Not required to obtain shutdown or cold shutdown. | 3 rd level Turbine Building | | 4/5 |
| GP-007 6.4.11.k | Place clearance on SDAFW Pump to prevent MDCT overflow | Not required to obtain shutdown or cold shutdown. | 1 st level Turbine Building | N/A | 4/5 |
| GP-007 6.4.11.l | Remove Condensate polishers from service as required | Not required to obtain shutdown or cold shutdown. | 1 st level Turbine Building | Condensate polisher Building | 4/5 |
| GP-007 6.4.13.b/c | Restore RHR train maintained subcooled | | REACTOR AUXILIARY BUILDING | E1/E2 Room, 1 st level Hallway, RHR Heat Exchanger Room | 4/5 |
| GP-007 8.4.14 | Remove Rod Control and RPI from service per OP-003 | Not required to obtain shutdown or cold shutdown. | REACTOR AUXILIARY BUILDING | Rod Drive MG Set Room, 2 nd level hallway | 4/5 |
| OP-915-1 | Makeup to CST and PWST | Makeup to Condensate Storage Tank is continuous due to Steam Generator Blow Down to flash tank | 1 st level Turbine Building | Makeup water Treatment (MWT) Room and Condensate Polisher | 1,2,3,4,5 |

ATTACHMENT 3

Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases

| In Plant Task - Procedure and Step | Step Action | Notes | Building | Room | Mode |
|------------------------------------|--|--|--|--|-----------|
| | | (loss), Condensate Polishers are typically not in operation however some equipment required for MWT is located in CP building. | | (CP) Building | |
| IAO rounds | Venting of Charging Pumps | Required to prevent gas intrusion concerns | REACTOR AUXILIARY BUILDING | Charging pump Room | 1,2,3,4,5 |
| OP-106 8.2.3 | Adjustment of SGBD flow rates | | 1 st and 2 nd level Turbine Building | | 1,2,3,4 |
| OP-910 8.4.3 | Stopping / Starting of Spent Fuel Pool Cooling pumps for temperature control | Required to maintain temperature in normal control band, frequency varies depending upon lake temperature | REACTOR AUXILIARY BUILDING | Spent Fuel Pump / Heat Exchanger Room | 1,2,3,4,5 |
| OP-301-2 6.2 | Cation Bed Demineralizer Operation | Frequency varies based on RCS Boron, required to maintain RCS chemistry | REACTOR AUXILIARY BUILDING | Primary Demineralizer Room | 1,2 |
| CP-PRI-207 | RCS Sampling / Chemical Control | Maintenance of RCS chemistry, Required to verify RCS boron concentrations to support SDM for shutdown and cooldown | REACTOR AUXILIARY BUILDING | Primary Sample Room / CCW Pump Room | 1,2,3,4,5 |
| CP-SEC-304 | Secondary Sampling / Chemical Control | Required to maintain secondary chemistry | 1 st level Turbine Building | Secondary Sample Room / Chemical Feed Room | 1,2,3,4 |

ATTACHMENT 3
Safe Operation & Shutdown Rooms/Areas Table R-2/H-2 Bases

Table R-2/H-2 Results

| Table R-2/H-2 Safe Operation & Shutdown Rooms/Areas | |
|---|----------------|
| Room/Area | Mode(s) |
| Reactor Auxiliary Building, 1 st level hallway | 1,2,3,4,5 |
| Reactor Auxiliary Building, 2 nd level hallway | 1,2,3,4,5 |
| Charging Pump Room | 1,2,3,4,5 |
| Component Cooling Water Pump Room | 1,2,3,4,5 |
| Primary Sample Room | 1,2,3,4,5 |
| Primary Demineralizer Room | 1,2,3 |
| Spent Fuel Pump / Heat Exchanger Room | 1,2,3,4,5 |
| Pipe Alley | 4 |
| RHR Heat Exchanger Room | 4 |
| RHR Pump Room entry area (access to RHR Pump CCW flow indication / control) | 4 |
| Boric Acid Batch Tank Room | 1,2,3,4,5 |
| Emergency Bus E1/E2 Room | 3,4,5 |
| Turbine Building 1 st Floor (includes Condensate Polisher, Makeup Water Treatment and Secondary Sample Room) | 1,2,3,4 |
| Turbine Building 2 nd Floor | 1,2,3,4 |
| Turbine Building 3 rd Floor | 1,3,4 |
| Containment Building | 3 |

RNP-RA/15-0034
Enclosure 5
34 Pages (including cover page)

Enclosure 5

SUPPORTING CALCULATION FOR HBRSEP2 EAL TABLE R-1, "EFFLUENT
MONITOR CLASSIFICATION THRESHOLDS"



Robinson Nuclear Plant (RNP)

Radiological Effluent EAL Values

**EP-EALCALC-RNP-1401
Revision 0**

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1. Purpose

The Robinson Nuclear Plant (RNP) Emergency Action Level (EAL) Technical Bases Manual contains background information, event declaration thresholds, bases and references for the site specific EAL and Fission Product Barrier (FPB) values used to implement the Nuclear Energy Institute (NEI) 99-01 Rev. 6 EAL guidance methodology. This calculation document provides additional technical detail specific to the derivation of the gaseous and liquid radiological effluent EAL values developed in accordance with the guidance in NEI 99-01 Rev. 6.

Documentation of the assumptions, calculations and results are provided for the RNP site specific Rx1 series EAL effluent monitor values associated the NEI 99-01 Rev 6 EALs listed below.

- NEI EAL AU1.1 (gaseous and liquid)
- NEI EAL AA1.1 (gaseous and liquid)
- NEI EAL AS1.1 (gaseous)
- NEI EAL AG1.1 (gaseous)

2. DEVELOPMENT METHODOLOGY AND BASES

2.1. Threshold Limits

2.1.1. RU1.1 Liquid Threshold Limits

Guidance Criteria

The RU1 Initiating Condition (IC) addresses a release of gaseous or liquid radioactivity greater than 2 times the Offsite Dose Calculation Manual (ODCM) limits for 60 minutes or longer.

RNP Bases

The ODCM Section 2.1 (Controls 2.2.1) limits for the concentration of radioactive liquid effluents released from the site to the unrestricted area are as follows:

- 10 times the effluent concentration (EC) levels of 10CFR20, Appendix B, Table 2
- 2.0E-04 $\mu\text{Ci/ml}$ for dissolved and entrained noble gases

The RU1.1 liquid effluent EAL threshold values will equate to 2 times the ODCM limit.

2.1.2. RU1.1 Gaseous Threshold Limits

Guidance Criteria

The RU1 Initiating Condition (IC) addresses a release of gaseous or liquid radioactivity greater than 2 times the Offsite Dose Calculation Manual (ODCM) limits for 60 minutes or longer.

RNP Bases

The ODCM Section 3.1 (Controls 3.2.1) limits for the concentration of radioactive gaseous effluents at the site boundary are as follows:

- Less than or equal to 500 mrem/yr to the whole body (Noble Gasses)
- Less than or equal to 3000 mrem/yr to the skin (Noble Gasses)
- Less than or equal to 1500 mrem/yr to any organ (I-131, I-133, tritium, and particulate with half-lives greater than 8 days)

Inhalation (internal organ) limits are not applicable for EAL threshold determination since the specified surveillance involves collection and analysis of composite samples. This after-the-fact assessment (individual uptake) could not be made in a timely manner conducive to accident classification.

The RU1.1 gaseous effluent EAL threshold values will equate to 2 times the ODCM limit for the lesser of the whole body or skin exposure pathways.

2.1.3. RA1.1 Liquid Threshold Limits

Guidance Criteria

The RA1 Initiating Condition (IC) addresses a release of radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.

This is based on values at 1% of the EPA Protective Action Guides (PAGs).

Per NEI 99-01, the effluent monitor readings should correspond to the above dose limits at the “site-specific dose receptor point” (consistent with the calculation methodology employed) for one hour of exposure.

RNP Bases

The liquid effluent limits are based on the water concentration values given in 10 CFR 20 Appendix B Table 2 Column 2 (see Section 2.1.1 above). The 10 CFR 20 values are equivalent to the radionuclide concentrations which, if ingested continuously over the course of a year, would produce a total effective dose equivalent of 0.05 rem (50 millirem). The EPA PAGs are based on a TEDE dose from immersion, inhalation and deposition. The 10 CFR 20 limits and the EPA limits do not represent the same type of exposure and thus cannot be compared on a one to one basis.

Additionally, significant dilution assumptions are incorporated in determining ODCM ingestion limits for liquid releases such that obtaining a dose of 10 mrem in one hour would require a discharge concentration above the effluent monitor threshold (ingestion of radioactivity from a liquid release at the site boundary is not practical).

Thus, the site specific EALs will not contain the RA1.1 liquid effluent monitor threshold value that equates to 1% of the EPA PAG. However, EALs RA1.3 and RA1.4 will remain applicable for liquid effluent releases that exceed the threshold based upon sample and field survey results.

2.1.4. RA1.1 Gaseous Threshold Limits

Guidance Criteria

The RA1 IC addresses a release of radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.

Per NEI 99-01, the effluent monitor readings are based on values at 1% of the EPA Protective Action Guides (PAGs) at the “site-specific dose receptor point” (consistent with the calculation methodology employed) for one hour of exposure.

RNP Bases

The gaseous effluent limits for RA1.1 are based on values that equate to an offsite dose greater than 10 mrem TEDE or 50 mrem CDE thyroid, which are 1% of the EPA PAGs.

2.1.5. RS1.1 Gaseous Threshold Limits

Guidance Criteria

The RS1 IC addresses a release of radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.

This is based on values at 10% of the EPA Protective Action Guides (PAGs) at the “site-specific dose receptor point” (consistent with the calculation methodology employed) for one hour of exposure.

RNP Bases

The gaseous effluent limits for RS1.1 are based on values that equate to an offsite dose greater than 100 mrem TEDE or 500 mrem CDE thyroid, which are 10% of the EPA PAGs.

2.1.6. RG1.1 Gaseous Threshold Limits

Guidance Criteria

The RG1 IC addresses a release of radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.

This is based on values at 100% of the EPA Protective Action Guides (PAGs) at the “site-specific dose receptor point” (consistent with the calculation methodology employed) for one hour of exposure.

RNP Bases

The gaseous effluent limits for RG1.1 are based on values that equate to an offsite dose greater than 1,000 mrem TEDE or 5,000 mrem CDE thyroid, which are 100% of the EPA PAGs.

2.2. Effluent Release Points

Note – All effluent release points assume a background reading of zero to conservatively account for all modes of operation applicable to the EALs.

2.2.1. Liquid Release Points

Guidance Criteria

Per NEI 99-01, the RU1 IC addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways (EAL #1) and planned batch releases from non-continuous release pathways (EAL #2).

Per NEI 99-01, the RA1 IC includes events or conditions involving a radiological release, whether gaseous or liquid, monitored or un-monitored. Classification based on effluent monitor readings assumes that a release path to the environment is established. If the 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 “site-specific monitor list and threshold values” should be determined with consideration of the selection of the appropriate installed gaseous and liquid effluent monitors.

RNP Bases

There are three liquid radwaste discharge points to the environment at RNP (ODCM 2.1 and Figure D-1):

1. Liquid Waste Disposal – R-18
2. Steam Generator Blowdown Monitor – R-19
3. Condensate Polisher Liquid Waste Monitor – R-37

Circulating water pump dilution flow rates range from 400,000 gpm with three unit 2 pumps running to 50,000 gpm with one unit 1 pump running (ODCM 2.1.1.1). Minimum dilution flow for the EAL threshold values for the release pathways above is assumed to be 50,000 gpm.

2.2.2. Gaseous Release Points

Guidance Criteria

Per NEI 99-01, the RU1 IC addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways (EAL #1) and planned batch releases from non-continuous release pathways (EAL #2).

Per NEI 99-01, the RA1 IC includes events or conditions involving a radiological release, whether gaseous or liquid, monitored or un-monitored. Classification based on effluent monitor readings assumes that a release path to the environment is established. If the 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.

Per NEI 99-01, the RS1 and RG1 ICs addresses monitored and un-monitored releases of gaseous radioactivity. 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.

The “site-specific monitor list and threshold values” should include the effluent monitors described in emergency plan and emergency dose assessment procedures.

RNP Bases

There are four gaseous effluent release points to the environment at RNP (ODCM Figure D-2):

1. Plant Vent – R-14C/D/E
2. Fuel Handling Basement Exhaust – R-20 (Low) and R-30 (High)

RNP fuel handling accidents are modeled through the plant vent for monitored events. Events involving the Waste Gas Tank (WGT) are modeled through the Fuel Handling Basement Exhaust via R-20/R-30.

3. E&RC Building Exhaust – R-22

4. Radwaste Building Exhaust – R-23

The E&RC Building Exhaust and Radwaste Building Exhaust pathways are not a source for normally occurring continuous radioactivity releases or for planned batch releases from non-continuous release pathways, and available activity is extremely low. The dose assessment model does not provide the capability to perform projections from these pathway monitors. Thus these pathways do not meet the NEI 99-01 criteria for use as an EAL threshold.

2.3. Source Term

2.3.1. RU1.1 Liquid Source Term

Guidance Criteria

NEI 99-01 does not provide specific guidance for AU1 liquid source term assumptions.

RNP Bases

The source term used for liquid effluent releases is Cs-134. Cs-134 has been selected based on it being the lowest effluent concentration value for any detectable radionuclide not known to be absent from the liquid effluent (ODCM 2.1).

2.3.2. RU1.1 Gaseous Source Term

Guidance Criteria

NEI 99-01 does not provide specific guidance for AU1 gaseous source term assumptions.

RNP Bases

The gaseous source term used for the Plant Vent release is based upon the NUREG-1940 Table 1-6 noble gas fraction of activity available at shutdown for fleet standardization.

The RU1.1 gaseous source term used for the Waste Gas Decay Tank release via the lower Fuel Handling Building is based upon the annual release concentration given in ODCM Table 3.1-1.

The RU1.1 source term is based on an activity mix that is limited to gaseous isotopes whereby all activity is assumed to be monitored as it is discharged.

2.3.3. RA1.1, RS1.1 and RG1.1 Gaseous Source Terms

Guidance Criteria

NEI 99-01 specifies that the calculation of monitor readings will require use of an assumed release isotopic mix; the selected mix should be the same for ICs AA1, AS1 and AG1.

RNP Bases

DEC utilizes a common plant vent RCS source term basis for fleet standardization. The source term utilized in the URI dose model provides the relative fractions and is taken from NUREG-1940 (referenced from URI Requirements Specification Appendix A Section A.1) with the release path 'I' selected to model a LOCA type event with fuel clad damage.

| | | | | | |
|------------|---|--------------------------------|--------------------------|-------------------|------------|
| RCS | Containment HUT < 2 hrs Sprays Off | Aux Bldg HUT < 2 hrs | Filter Working | Plant Vent | Env |
|------------|---|--------------------------------|--------------------------|-------------------|------------|

The RA/S/G1.1 source term utilized in the URI dose model for the pathway from the Waste Gas Decay Tank via the lower Fuel Handling Building is taken from FSAR section 15.7.1 (referenced from URI Requirements Specification Appendix A Section A.1) with the release path 'T' selected to model a tank rupture event.

| | | | |
|------------|---------------------------|------------------|------------|
| WGT | FHB HUT < 2 hrs | Lower FHB | Env |
|------------|---------------------------|------------------|------------|

No credit is taken for source term decay. The start of release time entered into URI is coincident with the time of reactor trip.

2.4. Release Duration

Guidance Criteria

Per NEI 99-01, the effluent monitor readings for RA1.1, RS1.1 and RG1.1 gaseous EAL threshold values should correspond to a dose at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.

RNP Bases

The effluent monitor readings for RA1.1, RS1.1 and RG1.1 gaseous EAL threshold values are calculated for a release duration of one hour.

2.5. Meteorology

Guidance Criteria

The effluent monitor readings should correspond to the applicable dose limit at the “site-specific dose receptor point.” The “site-specific dose receptor point” is the distance(s) and/or locations used by the licensee to distinguish between on-site and offsite doses. The selected distance(s) and/or locations should reflect the content of the emergency plan, and the procedural methodology used to determine offsite doses and protective action recommendations. This is typically the boundary of the Owner Controlled Area.

Monitor readings will be calculated using a set of assumed meteorological data or atmospheric dispersion factors; the data or factors selected for use should be the same for ICs AA1, AS1 and AG1.

RNP Bases

The site specific meteorology used for the calculation of monitor readings is based on selections and inputs for the URI dose assessment model as documented below.

2.5.1. Wind Speed and Stability Class (Median WS and stability memo – see Attachment 1)

The median meteorology values used to develop the EAL thresholds are as follows:

Median Wind Speed 4.4 mph
Stability Class (A-G) D

2.5.2. Wind Direction (ODCM 3.3.1 and Table 3.3-2)

The RNP “site-specific dose receptor point” utilized in the derivation of the EAL effluent release thresholds has been established as the closest on-land site boundary line, which is in the SSE (wind direction from 338°) sector at 0.26 miles for ground and mixed mode release levels.

2.5.3. Other Parameters

No precipitation is assumed to occur for the duration of the release and plume transport across the EPZ.

3. DESIGN INPUTS

3.1. General Constants and Conversion Factors

3.1.1. 472 cc/sec per cfm

3.1.2. 10^6 μ Ci per Ci

3.2. Liquid Effluent

3.2.1. Liquid Effluent Monitor Ranges (EPEOF-06 Att 10.10 Table 2)

Liquid Waste Disposal (R-18) 10^1 - 10^6 cpm

Steam Generator Blowdown (R-19) 10^1 - 10^7 cpm

Condensate Polisher Pump (R-37) 10^1 - 10^7 cpm

3.2.2. Liquid Effluent Dilution Flow (*F*)

Liquid Effluent Dilution Flow (ODCM 2.1.1.1) 5.00E+04 gpm

3.2.3. Liquid Effluent Source Flow (*f*)

Liquid Waste Disposal (ODCM 2.1.1.1) 60 gpm

Steam Generator Blowdown (ODCM 2.1.1.1) 160 gpm

Condensate Polisher pump (ODCM 2.1.1.1) 300 gpm

3.2.4. Recirculation Factor (σ)

The recirculation factor accounts for the fraction of discharged water reused by the station. The RNP ODCM liquid effluent setpoint calculation does not account for a recirculation factor, thus a value of 1.0 is used.

3.2.5. 10CFR20 Appendix B, Table 2, Column 2 Source Term Limit (*EC_i*)

Cs-134 $9.0\text{E-}07$ μ Ci/ml

The Cs-134 10CFR20 Appendix B, Table 2, Column 2 limit is $9.0\text{E-}07$ μ Ci/ml.

3.2.6. Cs-137 Correlation Factor (CF_i)

The liquid effluent monitor Cs-137 correlation factor converts the release concentration in $\mu\text{Ci/ml}$ to effluent monitor to cpm. The Cs-137 correlation factor is as follows:

Note –The lower of the R-19 correlation factors is used to represent the group of monitors. The conservative difference is less than 10% from the results using the highest correlation factor.

| | |
|-----------------------------------|---------------------------------|
| R-18 (Curve 6.6 11/06/90) | 2.72E+08 cpm/ $\mu\text{Ci/ml}$ |
| R-19A (Curve 6.7A 01/21/91) | 1.35E+08 cpm/ $\mu\text{Ci/ml}$ |
| R-19B (Curve 6.7B 01/21/91) | 1.32E+08 cpm/ $\mu\text{Ci/ml}$ |
| R-19C (Curve 6.7C 01/21/91) | 1.23E+08 cpm/ $\mu\text{Ci/ml}$ |
| R-37 (Curve 6.14 12/03/91) | 1.40E+08 cpm/ $\mu\text{Ci/ml}$ |

3.3. Gaseous Effluent

3.3.1. Gaseous Effluent Monitor Ranges (EPEOF-06 Att 10.10 Table 1)

| | |
|---|--------------------------|
| Plant Vent (L) – R-14C | $10^1\text{-}10^6$ cpm |
| Plant Vent (M) – R-14D | $10^1\text{-}10^6$ cpm |
| Plant Vent (H) – R-14E | $10^1\text{-}10^6$ cpm |
| Lower Fuel Handling Building (L) – R-20 | $10^1\text{-}10^7$ cpm |
| Lower Fuel Handling Building (H) – R-30..... | $10^0\text{-}10^5$ mR/hr |

3.3.2. Gaseous Effluent Source Flow (f)

| | |
|---|--------------|
| Plant Vent (ODCM 3.1.1.4)..... | 6.06E+04 cfm |
| Lower Fuel Handling Building (ODCM 3.1.1.4) | 1.02E+04 cfm |

3.3.3. RU1.1 Dispersion Factor (X/Q)

| | |
|---|---------------------------|
| Ground Level Dispersion Factor (ODCM App A Table A-1) | 8.08E-05 sec/m^3 |
| Mixed Mode Dispersion Factor (ODCM App A Table A-10) | 9.94E-07 sec/m^3 |

3.3.4. RU1.1 Source Term Fraction (S_i)

NUREG-1940 Table 1-6 noble gas fraction of activity available at shutdown is the fleet standard source term used for events that originate from the RCS and are release from the plant vent.

RNP source term concentrations for site specific release points other than the plant vent are taken from the annual release concentrations in ODCM Table 3.1-1.

| | Plant Vent (unitless) | Lower FHB (unitless) |
|----------------|----------------------------------|---------------------------------|
| Kr-83m | 1.83E-02 | 0.00E+00 |
| Kr-85 | 1.70E-03 | 8.00E-01 |
| Kr-85m | 3.71E-02 | 0.00E+00 |
| Kr-87 | 7.40E-02 | 0.00E+00 |
| Kr-88 | 1.02E-01 | 0.00E+00 |
| Xe-131m | 2.20E-03 | 4.50E-02 |
| Xe-133 | 3.26E-01 | 1.55E-01 |
| Xe-133m | 1.03E-02 | 0.00E+00 |
| Xe-135 | 8.54E-02 | 0.00E+00 |
| Xe-135m | 6.90E-02 | 0.00E+00 |
| Xe-138 | 2.74E-01 | 0.00E+00 |
| | 1.00E+00 | 1.00E+00 |

3.3.5. ODCM Dose Factors (Regulatory Guide 1.109 Table B-1)

Note – RG1.109 values converted from mRem/yr per pCi/m³ to mRem/yr per μ Ci/m³.

| | Total Body Dose Factor Ki (mRem/yr per uCi/m³) | Skin Beta Dose Factor Li (mRem/yr per uCi/m³) | Gamma Air Dose Factor Mi (mRad/yr per uCi/m³) |
|----------------|--|---|---|
| Kr-83m | 7.56E-02 | 0.00E+00 | 1.93E+01 |
| Kr-85 | 1.61E+01 | 1.34E+03 | 1.72E+01 |
| Kr-85m | 1.17E+03 | 1.46E+03 | 1.23E+03 |
| Kr-87 | 5.92E+03 | 9.73E+03 | 6.17E+03 |
| Kr-88 | 1.47E+04 | 2.37E+03 | 1.52E+04 |
| Xe-131m | 9.15E+01 | 4.76E+02 | 1.56E+02 |
| Xe-133 | 2.94E+02 | 3.06E+02 | 3.53E+02 |
| Xe-133m | 2.51E+02 | 9.94E+02 | 3.27E+02 |
| Xe-135 | 1.81E+03 | 1.86E+03 | 1.92E+03 |
| Xe-135m | 3.12E+03 | 7.11E+02 | 3.36E+03 |
| Xe-138 | 8.83E+03 | 4.13E+03 | 9.21E+03 |

3.3.6. RU1.1 Correlation Factor (CF)

The gaseous effluent monitor correlation factor converts the release concentration in $\mu\text{Ci/ml}$ to effluent monitor to cpm.

R-14C (Curve 6.2C Rev 207)2.99E+07 cpm/ $\mu\text{Ci/ml}$

R-20 (Curve 6.8 10/31/90)6.82E+07 cpm/ $\mu\text{Ci/ml}$

4. Calculations

4.1. RU1.1 Liquid Release

4.1.1. ODCM Liquid Release Limit

$$C_i \leq \frac{(F + f) \times (10 \times EC_i)}{\sigma \times f} \qquad SP \leq \sum_i (C_i \times CF) + bkg$$

Where:

| | |
|----------------------------|--|
| C_i | concentration of radionuclide 'i' in the liquid effluent (μCi/ml) – this is considered the ODCM limit for EAL purposes |
| F | dilution flow (gpm) |
| f | undiluted flow from the source of the release (gpm) |
| 10 | TS multiplier – component of ODCM Limit (see definition) |
| EC_i | concentration of radionuclide 'i' from 10CFR20, Appendix B, Table 2, Column 2 (μCi/ml) |
| σ | most restrictive recirculation factor at equilibrium (unitless) |
| SP | radiation monitor setpoint equivalent to the ODCM limit (cpm) |
| CF | radiation monitor correlation factor (cpm/μCi/ml) |
| bkg | background reading for the radiation monitor (cpm) |

4.1.2. RU1.1 Liquid Release EAL Threshold

$$RU1.1 = 2 \left(\sum_i (C_i \times CF) \right) + bkg$$

See Attachment 2 for the spreadsheet calculations that develop the RU1.1 liquid effluent EAL threshold values for each applicable monitor.

4.2. RU1.1 Gaseous Release

4.2.1. ODCM Gaseous Release Limit

$$SP_{\text{total body}} (\text{cpm}) = \left(\frac{500}{472 \times f \times \frac{X}{Q} \times \sum_i (S_i \times K_i)} \right) \times CF + bkg$$

$$SP_{\text{skin}} (\text{cpm}) = \left(\frac{3000}{472 \times f \times \frac{X}{Q} \times \sum_i (S_i \times (L_i + 1.1M_i))} \right) \times CF + bkg$$

Where:

| | |
|---|--|
| 500/3000 | ODCM Limit – 500 total body or 3000 skin (mrem/yr) |
| 472 | conversion factor (cc/ft ³ per sec/min) |
| f | vent flow (cfm) |
| X/Q | annual average meteorological dispersion to the controlling site boundary location (sec/m ³) |
| S_i | isotopic fraction of the mix activity released (unitless) |
| K_i | total body dose factor (mrem/yr per μCi/m ³) |
| L_i + 1.1M_i | skin dose factor (mrem/yr per μCi/m ³) |
| CF | radiation monitor correlation factor (cpm per μCi/ml) |
| bkg | background reading for the radiation monitor (cpm) |

4.2.2. RU1.1 Gaseous Release EAL Threshold

RU1.1 is two times the lesser of the calculated total body or skin value plus background.

See Attachment 3 for the spreadsheet calculations that develop the RU1.1 gaseous effluent EAL threshold values for each applicable monitor.

4.3. RA1.1, RS1.1 and RG1.1 Gaseous Release

The RA1.1, RS1.1 and RG1.1 gaseous release EAL threshold are developed using the URI site specific dose assessment models with the inputs described in Section 2 above.

Refer to Attachment 4 for the results of the URI gaseous effluent EAL threshold calculations.

5. Conclusion

| | Release Point | Monitor | GE | SAE | Alert | UE |
|--------|----------------------------------|---------|---------------|-----------------|-----------------|---------------|
| Gas | Plant Vent (H) | R-14E | 3.31E+3 (cpm) | 3.40E+2 (cpm) | 4.30E+1 (cpm) | N/A |
| | Plant Vent (M) | R-14D | 6.38E+5 (cpm) | 6.38E+4 (cpm) | 6.38E+3 (cpm) | N/A |
| | Plant Vent (L) | R-14C | N/A | N/A | N/A | 2.16E+5 (cpm) |
| | Lower FHB (H) | R-30 | N/A | 2.60E+4 (mR/hr) | 2.60E+3 (mR/hr) | N/A |
| | Lower FHB (L) | R-20 | N/A | N/A | N/A | 8.60E+5 (cpm) |
| Liquid | Liquid Waste Disposal | R-18 | N/A | N/A | N/A | 4.08E+6 (cpm) |
| | Steam Generator Blowdown | R-19 | N/A | N/A | N/A | 6.94E+5 (cpm) |
| | Condensate Polisher Liquid Waste | R-37 | N/A | N/A | N/A | 4.23E+5 (cpm) |

6. References

- 6.1. NEI 99-01 R6, Methodology for Development of Emergency Action Levels, November 2012
- 6.2. NUREG-1940, RASCAL 4: Description of Models and Methods, December 2012
- 6.3. HB Robinson Steam Electric Plant Unit 2 Offsite Dose Calculation Manual (ODCM), Revision 33
- 6.4. Unified RASCAL Interface Requirements Specification, Robinson, Version 2
- 6.5. Station Curve Book, Revision 214
- 6.6. EMP-023 Attachment 10.1, Release Point Description, Revision 61
- 6.7. EPEOF-06 Attachment 10.10, Radiation Monitor Information, Revision 10
- 6.8. Memo: Median Wind Speed and Stability Values at Duke Energy Nuclear Sites, 06/19/14

Date: June 19, 2014

To: Caryl Ingram, NGO-EP

From: Stanton Lanham, Meteorology - Environmental Services
Marsha Kinley, Meteorology - Environmental Services

Subject: Median Wind Speed and Stability Values at Duke Energy Nuclear Sites

1.0 Overview

Data from the most recent full five years (2009-2013) was used to calculate the median wind speed (WS), vertical temperature gradient (Delta-T), and stability class at each of the Duke Energy nuclear sites in the Carolinas. Upper level winds were used at Brunswick. All other sites use the lower level. Singular median values for WS, Delta-T, and stability class from all wind direction sectors are given in Table 1. NEI 99-01 Rev. 6 does not provide any guidance on selection of default meteorological conditions.

- These median values are irrespective of season or time of day, so the difference between the median values and actual meteorological conditions could be large.
- Also note that the median Delta-T values are in normalized units of (deg C/100m), and would need to be converted to reflect actual sensor separation distance on a tower, if needed.

Table 2.1 through Table 2.6 contains sector-specific median values of Wind Speed, Delta-T and Stability Class for each of the 16 directional sectors. This information provides more site-specific characteristics, similar to what would have been evaluated for the previous Rev. 4 of NEI-99-01 guidance. In addition, the most frequent sector from which the wind is blowing at each site for the five year period is also indicated in these tables.

Table 1 Median Values from Years 2009-2013

| | Median WS (mph) | Median Delta-T (C/100m) ** | Stability Class |
|-----------|-----------------|----------------------------|-----------------|
| DEC Sites | | | |
| CNS | 4.8 | -0.7 | D |
| MNS | 6 | -0.9 | D |
| ONS | 3.7 | -0.78 | D |
| DEP Sites | | | |
| BNP* | 13.4 | -0.71 | D |
| HNP | 3.5 | -0.51 | D |
| RNP | 4.4 | -0.84 | D |

* Upper level winds are used at BNP. All other sites use lower level winds.

**Note: Delta-T values listed are in degs C/100 m. The units may need to be converted if actual delta-T based on tower-specific separation distances are required.

2.0 Data

The data presented represents the median of the entire five-year span at each site (Table 1), as well as the overall medians broken down by directional sector (Tables 2.1 - 2.6). Each value represents the middle of the dataset, with 50% of values above the median, and 50% of values below the median.

Data for the Legacy Duke Energy sites was obtained from the Duke's Environmental Monitoring "Ambient Administration" archive, which contains validated hourly meteorological data. Data for the Legacy Progress sites was obtained from hourly meteorological data files provided by the vendor (Murray and Trettel), and has undergone their data review/QA process. The five-year analysis results presented here were determined independently of previous studies, however comparison to the Annual Effluent reports (2013 MET) for all sites showed good agreement with the values presented in Table 1. The sector-specific median values (Tables 2.1 through 2.6) had not been investigated previously.

Legacy Duke Sites (DEC):

Table 2.1 Catawba Nuclear: 5-year Lower Level Medians by Sector

| Sector | Median WS (mph) | Median Delta-T (C/100m) | Stability Class |
|--------|-----------------|-------------------------|-----------------|
| N | 7.4 | -1.08 | D |
| NNE | 8.7 | -1.3 | D |
| NE | 9 | -1.2 | D |
| ENE | 6.1 | -1.06 | D |
| E | 4.6 | -0.94 | D |
| ESE | 4.4 | -0.9 | D |
| SE | 4.8 | -0.8 | D |
| SSE | 4.4 | -0.76 | D |
| S* | 3.9 | -0.36 | E |
| SSW | 4.1 | -0.66 | D |
| SW | 3.8 | -0.7 | D |
| WSW | 3.4 | -0.4 | E |
| W | 3.6 | 0 | E |
| WNW | 4 | 0 | E |
| NW | 4.4 | 0 | E |
| NNW | 5.1 | 0 | E |

* Most frequent CNS wind direction (2009-2013): from South

Table 2.2 McGuire Nuclear: 5-year Lower Level Medians by Sector

| Sector | Median WS (mph) | Median Delta-T (C/100m) | Stability Class |
|--------|-----------------|-------------------------|-----------------|
| N | 6.9 | -1.18 | D |
| NNE | 7 | -1.16 | D |
| NE | 7.8 | -1.06 | D |
| ENE | 6.6 | -1.02 | D |
| E | 6.2 | -0.88 | D |
| ESE | 5.5 | -0.88 | D |
| SE | 5.1 | -0.68 | D |
| SSE | 4.2 | -0.42 | E |
| S | 4.6 | -0.12 | E |
| SSW | 5 | -0.14 | E |
| SW* | 6.3 | -0.72 | D |
| WSW | 5.2 | -0.74 | D |
| W | 4.9 | -0.76 | D |
| WNW | 6.3 | -0.92 | D |
| NW | 8.5 | -1.06 | D |
| NNW | 9.1 | -1.16 | D |

* Most frequent MNS wind direction (2009-2013): from SW

Table 2.3 Oconee Nuclear: 5-year Lower Level Medians by Sector

| Sector | Median WS (mph) | Median Delta-T (C/100m) | Stability Class |
|--------|-----------------|-------------------------|-----------------|
| N | 2.5 | -0.44 | E |
| NNE | 2.8 | -0.58 | D |
| NE | 3.9 | -0.84 | D |
| ENE | 4.6 | -0.88 | D |
| E | 3.7 | -0.72 | D |
| ESE | 3.2 | -0.4 | E |
| SE | 3.3 | -0.42 | E |
| SSE | 3.3 | -0.5 | D |
| S | 3.4 | -0.68 | D |
| SSW | 4.6 | -1.2 | D |
| SW* | 5 | -1.32 | D |
| WSW | 4.8 | -1.06 | D |
| W | 3.6 | -0.8 | D |
| WNW | 2.8 | -0.46 | E |
| NW | 2.7 | -0.2 | E |
| NNW | 2.5 | -0.42 | E |

* Most frequent ONS wind direction (2009-2013): from SW

Legacy Progress Sites (DEP):

Table 2.4 Brunswick Nuclear: 5-year Upper Level Medians by Sector

| Sector | Median WS (mph) | Median Delta-T (C/100m) | Stability Class |
|--------|-----------------|-------------------------|-----------------|
| N | 14 | -0.71 | D |
| NNE | 14.7 | -0.68 | D |
| NE | 14.2 | -0.69 | D |
| ENE | 13.6 | -0.81 | D |
| E | 11.2 | -0.79 | D |
| ESE | 9.5 | -0.635 | D |
| SE | 9 | -0.67 | D |
| SSE | 9.5 | -0.46 | E |
| S | 11.4 | -0.34 | E |
| SSW | 13.6 | -0.79 | D |
| SW* | 16.2 | -0.95 | D |
| WSW | 14.2 | -0.74 | D |
| W | 9.6 | -0.24 | E |
| WNW | 14.3 | -0.28 | E |
| NW | 14.4 | -0.44 | E |
| NNW | 15 | -0.7 | D |

* Most frequent BNP wind direction (2009-2013): from SW

Table 2.5 Harris Nuclear: 5-year Lower Level Medians by Sector

| Sector | Median WS (mph) | Median Delta-T (C/100m) | Stability Class |
|--------|-----------------|-------------------------|-----------------|
| N | 3.3 | -0.35 | E |
| NNE* | 3.1 | -0.26 | E |
| NE | 1.6 | 0.92 | E |
| ENE | 2.1 | 0.26 | E |
| E | 2.2 | -0.07 | E |
| ESE | 2.6 | -0.39 | E |
| SE | 2.9 | -0.49 | E |
| SSE | 3.4 | -0.59 | D |
| S | 4.2 | -0.64 | D |
| SSW | 4.7 | -0.58 | D |
| SW | 4.7 | -0.64 | D |
| WSW | 4.6 | -0.86 | D |
| W | 3.7 | -0.68 | D |
| WNW | 4.2 | -0.74 | D |
| NW | 4.1 | -0.805 | D |
| NNW | 3.5 | -0.55 | D |

* Most frequent HNP wind direction (2009-2013): from NNE

Table 2.6 Robinson Nuclear: 5-year Lower Level Medians by Sector

| Sector | Median WS (mph) | Median Delta-T (C/100m) | Stability Class |
|--------|-----------------|-------------------------|-----------------|
| N* | 5.8 | -1.03 | D |
| NNE | 5.2 | -1.09 | D |
| NE | 4 | -1.11 | D |
| ENE | 3.8 | -1.14 | D |
| E | 3.6 | -1.2 | D |
| ESE | 3.3 | -1.28 | D |
| SE | 3.5 | -1.12 | D |
| SSE | 4.2 | -0.69 | D |
| S | 4.7 | -0.6 | D |
| SSW | 4.6 | -0.68 | D |
| SW | 4.6 | -0.83 | D |
| WSW | 4 | -0.71 | D |
| W | 3.9 | -0.59 | D |
| WNW | 3.9 | -0.47 | E |
| NW | 4.1 | 0.28 | E |
| NNW | 4.7 | 0.31 | E |

* Most frequent RNP wind direction (2009-2013): from North

3.0 Discussion and Conclusion

The median wind speed data presented in Table 1 compared to Tables 2.1 through 2.6 indicates typically varying conditions, depending on the directional sectors at each site. The overall median wind speed at a site (3-6 mph) is in the middle of the wider range of the sector-specific medians (1-9 mph). The singular median values sometime match well with the sector-specific median conditions of the most frequent directional sector, but can also be entirely different from the median of the most frequent wind direction sector. These differences span from potentially lower wind speeds which would be conservative for dose (i.e. Brunswick), to potentially higher wind speeds which would be non-conservative for dose (i.e. Catawba).

- Thus, the median values of wind speed should only be used for dose assessment as a last resort, when actual meteorological data is not available, or dose calculation is for some reason impaired.

The median Stability Class is generally neutral (class D), but varies between D and E (slightly more stable) in the sector-specific tables (Tables 2.1 through 2.6). These median values are typical of daytime conditions, with a thermally mixed boundary layer.

- Thus, the median stability class should only be used when there is no concern about actual time of day, seasonal variances, or extreme weather events.

| Monitor | Dilution Flow (F) | Undiluted Flow (f) | Recirculation Factor (σ) | Correlation Factor (CFi) | Maximum Allowable Concentration - Ci ($\mu\text{Ci/ml}$) | Radiation Monitor Setpoint - SP (cpm) | RU1.1 EAL Threshold Value (cpm) |
|---------|-------------------|--------------------|-----------------------------------|--------------------------|--|---------------------------------------|---------------------------------|
| R-18 | 5.00E+04 | 60 | 1 | 2.72E+08 | 7.51E-03 | 2.04E+06 | 4.08E+06 |
| R-19 | 5.00E+04 | 160 | 1 | 1.23E+08 | 2.82E-03 | 3.47E+05 | 6.94E+05 |
| R-37 | 5.00E+04 | 300 | 1 | 1.40E+08 | 1.51E-03 | 2.11E+05 | 4.23E+05 |

Cs-134 10CFR20 Limit - ECI ($\mu\text{Ci/ml}$): 9.0E-07

TS Multiplier: 1.0E+01

Background (cpm): 0

| | PV Fraction - Si | WGDT Fraction - Si | Total Body Dose Factor - Ki (mRem/yr per $\mu\text{Ci}/\text{m}^3$) | Skin Beta Dose Factor - Li (mRem/yr per $\mu\text{Ci}/\text{m}^3$) | Gamma Air Dose Factor - Mi (mRad/yr per $\mu\text{Ci}/\text{m}^3$) | Plant Vent | | Lower FHB | |
|----------------|------------------|--------------------|---|--|--|---|---|---|---|
| | | | | | | Si x Ki (mRem/yr per $\mu\text{Ci}/\text{m}^3$) | Si x (Li + 1.1Mi) (mRem/yr per $\mu\text{Ci}/\text{m}^3$) | Si x Ki (mRem/yr per $\mu\text{Ci}/\text{m}^3$) | Si x (Li + 1.1Mi) (mRem/yr per $\mu\text{Ci}/\text{m}^3$) |
| Kr-83m | 1.83E-02 | 0.00E+00 | 7.56E-02 | 0.00E+00 | 1.93E+01 | 1.38E-03 | 3.89E-01 | 0.00E+00 | 0.00E+00 |
| Kr-85 | 1.70E-03 | 8.00E-01 | 1.61E+01 | 1.34E+03 | 1.72E+01 | 2.74E-02 | 2.31E+00 | 1.29E+01 | 1.09E+03 |
| Kr-85m | 3.71E-02 | 0.00E+00 | 1.17E+03 | 1.46E+03 | 1.23E+03 | 4.34E+01 | 1.04E+02 | 0.00E+00 | 0.00E+00 |
| Kr-87 | 7.40E-02 | 0.00E+00 | 5.92E+03 | 9.73E+03 | 6.17E+03 | 4.38E+02 | 1.22E+03 | 0.00E+00 | 0.00E+00 |
| Kr-88 | 1.02E-01 | 0.00E+00 | 1.47E+04 | 2.37E+03 | 1.52E+04 | 1.50E+03 | 1.95E+03 | 0.00E+00 | 0.00E+00 |
| Xe-131m | 2.20E-03 | 4.50E-02 | 9.15E+01 | 4.76E+02 | 1.56E+02 | 2.01E-01 | 1.42E+00 | 4.12E+00 | 2.91E+01 |
| Xe-133 | 3.26E-01 | 1.55E-01 | 2.94E+02 | 3.06E+02 | 3.53E+02 | 9.59E+01 | 2.26E+02 | 4.56E+01 | 1.08E+02 |
| Xe-133m | 1.03E-02 | 0.00E+00 | 2.51E+02 | 9.94E+02 | 3.27E+02 | 2.59E+00 | 1.39E+01 | 0.00E+00 | 0.00E+00 |
| Xe-135 | 8.54E-02 | 0.00E+00 | 1.81E+03 | 1.86E+03 | 1.92E+03 | 1.55E+02 | 3.39E+02 | 0.00E+00 | 0.00E+00 |
| Xe-135m | 6.90E-02 | 0.00E+00 | 3.12E+03 | 7.11E+02 | 3.36E+03 | 2.15E+02 | 3.04E+02 | 0.00E+00 | 0.00E+00 |
| Xe-138 | 2.74E-01 | 0.00E+00 | 8.83E+03 | 4.13E+03 | 9.21E+03 | 2.42E+03 | 3.90E+03 | 0.00E+00 | 0.00E+00 |
| | 1.00E+00 | 1.00E+00 | | | | 4.87E+03 | 8.07E+03 | 6.26E+01 | 1.22E+03 |

| | | | | | | |
|---|--|------|--|--|----------|----------|
| Conversion Factor (cc/ft ³ per sec/min): | | 472 | Vent Flow (cfm): | | R-14C | R-20 |
| Total Body Dose Rate Limit (mRem/yr): | | 500 | X/Q (sec/m ³): | | 6.06E+04 | 1.02E+04 |
| Skin Dose Rate Limit (mRem/yr): | | 3000 | Correlation Factor (cpm/ $\mu\text{Ci}/\text{ml}$): | | 9.94E-07 | 8.08E-05 |
| Background (cpm): | | 0 | Total Body ODCM Limit ($\mu\text{Ci}/\text{ml}$): | | 2.99E+07 | 6.82E+07 |
| | | | Skin ODCM Limit ($\mu\text{Ci}/\text{ml}$): | | 3.61E-03 | 2.05E-02 |
| | | | Total Body ODCM Limit (cpm): | | 1.31E-02 | 6.30E-03 |
| | | | Skin ODCM Limit (cpm): | | 1.08E+05 | 1.40E+06 |
| | | | 2x ODCM Limit (cpm): | | 3.91E+05 | 4.30E+05 |
| | | | | | 2.16E+05 | 8.60E+05 |

Dose Assessment

Robinson

Tuesday, January 13, 2015 18:05

Method: Detailed Assessment - Monitored Release

Release Pathway: (I) <RCS> <Containment> <Aux Bldg> <Filters> <Plant Vent> <Env>

PRF: 1.60E-03

Containment HUT: = < 2 Hours

Cont Sprays: = OFF

Purge Filter: = N/A

Aux/Fuel Bldg HUT: = < 2 Hours

Aux/Fuel Filter: = Working

Steam Gen: = N/A

Turb Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 0:00

Release Duration (hh:mm): 1:00

ETE (hh:mm): [N/A]

Upper

Wind: From 338° @ 4.4 mph

Stability Class: D

Precipitation: None

Monitor: R14D Plnt Vnt mid

Readings: 6.38E+03 cpm

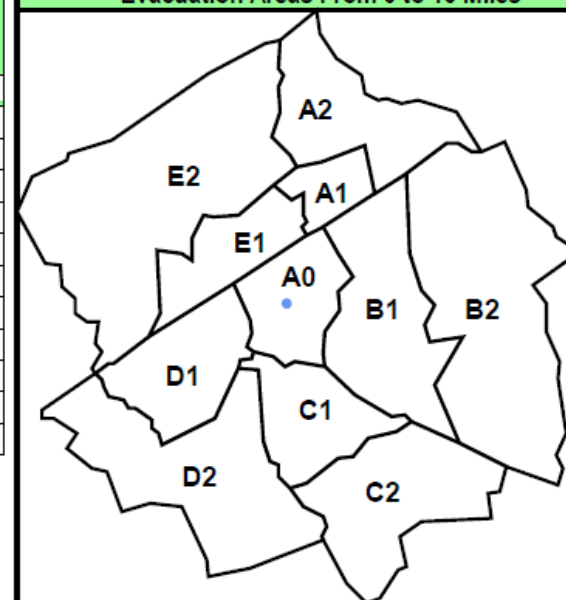
Flowrate: 60600 CFM

| Distance (Miles) | Exposure Rate (mR/hr) | External Plume DDE (mRem) | Inhalation CEDE (mRem) | Deposition Ground DDE (mRem) | TEDE (mRem) | CDE Thyroid (mRem) |
|---------------------|-----------------------------|---------------------------------|------------------------------|------------------------------------|----------------|--------------------------|
| S.B. | 1.25E+01 | 8.71E+00 | 1.00E+00 | 3.17E-01 | 1.00E+01 | 2.54E+01 |
| 0.3 | 1.14E+01 | 7.92E+00 | 8.40E-01 | 2.65E-01 | 9.03E+00 | 2.12E+01 |
| 0.5 | 7.36E+00 | 5.00E+00 | 5.68E-01 | 3.57E-01 | 5.93E+00 | 1.15E+01 |
| 0.7 | 5.28E+00 | 3.51E+00 | 4.32E-01 | 3.33E-01 | 4.27E+00 | 7.60E+00 |
| 1.0 | 3.40E+00 | 2.23E+00 | 3.04E-01 | 2.65E-01 | 2.80E+00 | 4.68E+00 |
| 1.5 | 1.85E+00 | 1.20E+00 | 1.88E-01 | 1.60E-01 | 1.55E+00 | 2.70E+00 |
| 2.0 | 1.42E+00 | 9.44E-01 | 1.29E-01 | 0.00E+00 | 1.07E+00 | 1.93E+00 |
| 3.0 | 1.20E+00 | 7.83E-01 | 0.00E+00 | 0.00E+00 | 7.83E-01 | 1.23E+00 |
| 4.0 | 1.13E+00 | 7.36E-01 | 0.00E+00 | 0.00E+00 | 7.36E-01 | 1.24E+00 |
| 5.0 | 8.72E-01 | 5.67E-01 | 0.00E+00 | 0.00E+00 | 5.67E-01 | 1.07E+00 |
| 7.0 | 5.64E-01 | 3.52E-01 | 0.00E+00 | 0.00E+00 | 3.52E-01 | 8.13E-01 |
| 10.0 | 2.78E-01 | 1.85E-01 | 0.00E+00 | 0.00E+00 | 1.85E-01 | 5.40E-01 |

Assessment Data Results Saved to File:

Robinson 10Miles Monitored Release 01132015 180550.URI7

Evacuation Areas From 0 to 10 Miles



No PAGs Exceeded

*** Classification: Validate against Emergency Action Levels ***

Release Rates (Ci / sec)

| | |
|-------------|------------------|
| Particulate | 2.86E-04 (0.0%) |
| Iodine | 7.00E-03 (0.2%) |
| Noble Gas | 2.88E+00 (99.7%) |

Reviewed By: _____

Dose Assessment

Robinson

Tuesday, January 13, 2015 18:06

Method: Detailed Assessment - Monitored Release

Release Pathway: (I) <RCS> <Containment> <Aux Bldg> <Filters> <Plant Vent> <Env>

PRF: 1.60E-03

Containment HUT: = < 2 Hours

Cont Sprays: = OFF

Purge Filter: = N/A

Aux/Fuel Bldg HUT: = < 2 Hours

Aux/Fuel Filter: = Working

Steam Gen: = N/A

Turb Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Upper

Time After S/D (hh:mm): 0:00

Wind: From 338° @ 4.4 mph

Release Duration (hh:mm): 1:00

ETE (hh:mm): [N/A]

Stability Class: D

Precipitation: None

Monitor: R14D Plnt Vnt mid

Readings: 6.38E+04 cpm

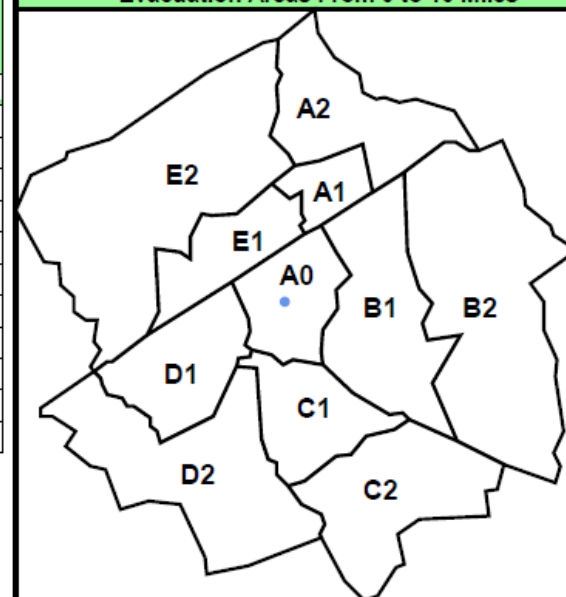
Flowrate: 60600 CFM

| Distance (Miles) | Exposure Rate (mR/hr) | External Plume DDE (mRem) | Inhalation CEDE (mRem) | Deposition Ground DDE (mRem) | TEDE (mRem) | CDE Thyroid (mRem) |
|---------------------|-----------------------------|---------------------------------|------------------------------|------------------------------------|----------------|--------------------------|
| S.B. | 1.25E+02 | 8.71E+01 | 1.00E+01 | 3.18E+00 | 1.00E+02 | 2.54E+02 |
| 0.3 | 1.14E+02 | 7.92E+01 | 8.40E+00 | 2.66E+00 | 9.03E+01 | 2.13E+02 |
| 0.5 | 7.36E+01 | 5.00E+01 | 5.68E+00 | 3.58E+00 | 5.93E+01 | 1.15E+02 |
| 0.7 | 5.28E+01 | 3.51E+01 | 4.32E+00 | 3.33E+00 | 4.28E+01 | 7.60E+01 |
| 1.0 | 3.41E+01 | 2.23E+01 | 3.05E+00 | 2.66E+00 | 2.80E+01 | 4.68E+01 |
| 1.5 | 1.85E+01 | 1.20E+01 | 1.88E+00 | 1.60E+00 | 1.55E+01 | 2.70E+01 |
| 2.0 | 1.42E+01 | 9.44E+00 | 1.29E+00 | 9.33E-01 | 1.17E+01 | 1.93E+01 |
| 3.0 | 1.20E+01 | 7.84E+00 | 8.54E-01 | 6.86E-01 | 9.38E+00 | 1.23E+01 |
| 4.0 | 1.13E+01 | 7.37E+00 | 8.56E-01 | 6.52E-01 | 8.88E+00 | 1.24E+01 |
| 5.0 | 8.72E+00 | 5.67E+00 | 7.18E-01 | 4.89E-01 | 6.87E+00 | 1.08E+01 |
| 7.0 | 5.64E+00 | 3.52E+00 | 5.05E-01 | 2.76E-01 | 4.31E+00 | 8.16E+00 |
| 10.0 | 2.78E+00 | 1.85E+00 | 3.06E-01 | 1.26E-01 | 2.28E+00 | 5.41E+00 |

Assessment Data Results Saved to File:

Robinson 10Miles Monitored Release 01132015 180629.URI7

Evacuation Areas From 0 to 10 Miles



No PAGs Exceeded

*** Classification: Site Area Emergency ***

Release Rates (Ci / sec)

| | |
|-------------|------------------|
| Particulate | 2.86E-03 (0.0%) |
| Iodine | 7.01E-02 (0.2%) |
| Noble Gas | 2.88E+01 (99.7%) |

Reviewed By: _____

Dose Assessment

Robinson

Tuesday, January 13, 2015 18:06

Method: Detailed Assessment - Monitored Release

Release Pathway: (I) <RCS> <Containment> <Aux Bldg> <Filters> <Plant Vent> <Env>

PRF: 1.60E-03

Containment HUT: = < 2 Hours

Cont Sprays: = OFF

Purge Filter: = N/A

Aux/Fuel Bldg HUT: = < 2 Hours

Aux/Fuel Filter: = Working

Steam Gen: = N/A

Turb Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Upper

Time After S/D (hh:mm): 0:00

Wind: From 338° @ 4.4 mph

Release Duration (hh:mm): 1:00

ETE (hh:mm): [N/A]

Stability Class: D

Precipitation: None

Monitor: R14D Plnt Vnt mid

Readings: 6.38E+05 cpm

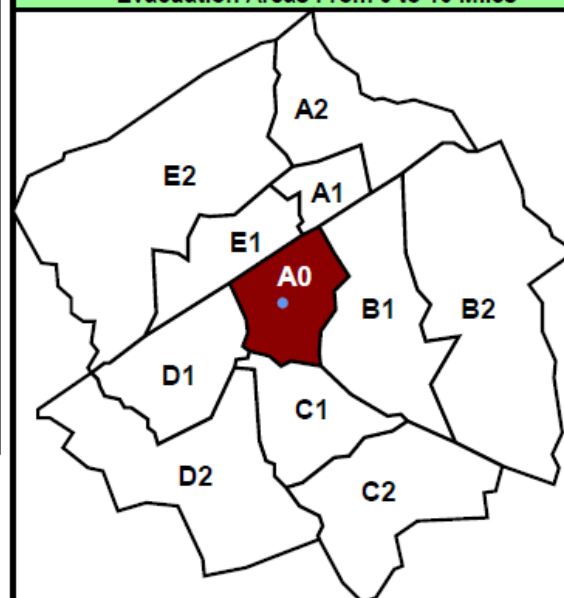
Flowrate: 60600 CFM

| Distance (Miles) | Exposure Rate (mR/hr) | External Plume DDE (mRem) | Inhalation CEDE (mRem) | Deposition Ground DDE (mRem) | TEDE (mRem) | CDE Thyroid (mRem) |
|---------------------|-----------------------------|---------------------------------|------------------------------|------------------------------------|----------------|--------------------------|
| S.B. | 1.25E+03 | 8.71E+02 | 1.00E+02 | 3.18E+01 | 1.00E+03 | 2.54E+03 |
| 0.3 | 1.14E+03 | 7.92E+02 | 8.40E+01 | 2.66E+01 | 9.03E+02 | 2.13E+03 |
| 0.5 | 7.36E+02 | 5.00E+02 | 5.72E+01 | 3.58E+01 | 5.93E+02 | 1.15E+03 |
| 0.7 | 5.28E+02 | 3.51E+02 | 4.32E+01 | 3.33E+01 | 4.28E+02 | 7.60E+02 |
| 1.0 | 3.41E+02 | 2.23E+02 | 3.05E+01 | 2.66E+01 | 2.80E+02 | 4.68E+02 |
| 1.5 | 1.85E+02 | 1.20E+02 | 1.88E+01 | 1.60E+01 | 1.55E+02 | 2.70E+02 |
| 2.0 | 1.42E+02 | 9.44E+01 | 1.29E+01 | 9.33E+00 | 1.17E+02 | 1.93E+02 |
| 3.0 | 1.20E+02 | 7.84E+01 | 8.54E+00 | 6.86E+00 | 9.38E+01 | 1.23E+02 |
| 4.0 | 1.13E+02 | 7.37E+01 | 8.56E+00 | 6.52E+00 | 8.88E+01 | 1.24E+02 |
| 5.0 | 8.72E+01 | 5.67E+01 | 7.18E+00 | 4.89E+00 | 6.87E+01 | 1.08E+02 |
| 7.0 | 5.64E+01 | 3.52E+01 | 5.05E+00 | 2.76E+00 | 4.31E+01 | 8.16E+01 |
| 10.0 | 2.78E+01 | 1.85E+01 | 3.06E+00 | 1.26E+00 | 2.28E+01 | 5.41E+01 |

Assessment Data Results Saved to File:

Robinson 10Miles Monitored Release 01132015 180658.URI7

Evacuation Areas From 0 to 10 Miles



PAGs Exceeded in Designated Areas

***** Classification: General Emergency *****

Release Rates (Ci / sec)

| | |
|-------------|------------------|
| Particulate | 2.86E-02 (0.0%) |
| Iodine | 7.01E-01 (0.2%) |
| Noble Gas | 2.88E+02 (99.7%) |

Reviewed By: _____

Dose Assessment

Robinson

Monday, March 09, 2015 08:26

Method: Detailed Assessment - Monitored Release

Release Pathway: (I) <RCS> <Containment> <Aux Bldg> <Filters> <Plant Vent> <Env>

PRF: 1.60E-03

Containment HUT: = < 2 Hours

Cont Sprays: = OFF

Purge Filter: = N/A

Aux/Fuel Bldg HUT: = < 2 Hours

Aux/Fuel Filter: = Working

Steam Gen: = N/A

Turb Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Upper

Time After S/D (hh:mm): 0:00

Wind: From 338° @ 4.4 mph

Release Duration (hh:mm): 1:00

ETE (hh:mm): [N/A]

Stability Class: D

Precipitation: None

Monitor: R14E Plnt Vnt HI

Readings: 4.30E+01 cpm

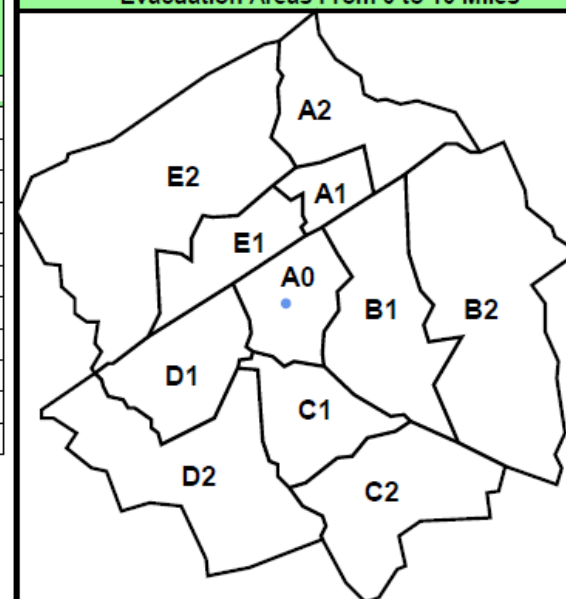
Flowrate: 60600 CFM

| Distance (Miles) | Exposure Rate (mR/hr) | External Plume DDE (mRem) | Inhalation CEDE (mRem) | Deposition Ground DDE (mRem) | TEDE (mRem) | CDE Thyroid (mRem) |
|---------------------|-----------------------------|---------------------------------|------------------------------|------------------------------------|----------------|--------------------------|
| S.B. | 1.24E+01 | 8.70E+00 | 1.00E+00 | 3.16E-01 | 1.00E+01 | 2.54E+01 |
| 0.3 | 1.13E+01 | 7.92E+00 | 8.40E-01 | 2.65E-01 | 9.02E+00 | 2.12E+01 |
| 0.5 | 7.32E+00 | 4.96E+00 | 5.68E-01 | 3.56E-01 | 5.88E+00 | 1.15E+01 |
| 0.7 | 5.24E+00 | 3.50E+00 | 4.32E-01 | 3.32E-01 | 4.26E+00 | 7.60E+00 |
| 1.0 | 3.40E+00 | 2.22E+00 | 3.04E-01 | 2.65E-01 | 2.79E+00 | 4.68E+00 |
| 1.5 | 1.84E+00 | 1.20E+00 | 1.88E-01 | 1.59E-01 | 1.54E+00 | 2.69E+00 |
| 2.0 | 1.42E+00 | 9.40E-01 | 1.29E-01 | 0.00E+00 | 1.07E+00 | 1.93E+00 |
| 3.0 | 1.19E+00 | 7.80E-01 | 0.00E+00 | 0.00E+00 | 7.80E-01 | 1.23E+00 |
| 4.0 | 1.12E+00 | 7.36E-01 | 0.00E+00 | 0.00E+00 | 7.36E-01 | 1.24E+00 |
| 5.0 | 8.68E-01 | 5.64E-01 | 0.00E+00 | 0.00E+00 | 5.64E-01 | 1.07E+00 |
| 7.0 | 5.60E-01 | 3.51E-01 | 0.00E+00 | 0.00E+00 | 3.51E-01 | 8.13E-01 |
| 10.0 | 2.77E-01 | 1.85E-01 | 0.00E+00 | 0.00E+00 | 1.85E-01 | 5.40E-01 |

Assessment Data Results Saved to File:

Robinson 10Miles Monitored Release 03092015 082642.URI7

Evacuation Areas From 0 to 10 Miles



No PAGs Exceeded

*** Classification: Validate against Emergency Action Levels ***

Release Rates (Ci / sec)

| | |
|-------------|------------------|
| Particulate | 2.85E-04 (0.0%) |
| Iodine | 6.99E-03 (0.2%) |
| Noble Gas | 2.87E+00 (99.7%) |

Reviewed By: _____

Dose Assessment

Robinson

Monday, March 09, 2015 08:29

Method: Detailed Assessment - Monitored Release

Release Pathway: (I) <RCS> <Containment> <Aux Bldg> <Filters> <Plant Vent> <Env>

PRF: 1.60E-03

Containment HUT: = < 2 Hours

Cont Sprays: = OFF

Purge Filter: = N/A

Aux/Fuel Bldg HUT: = < 2 Hours

Aux/Fuel Filter: = Working

Steam Gen: = N/A

Turb Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Upper

Time After S/D (hh:mm): 0:00

Wind: From 338° @ 4.4 mph

Release Duration (hh:mm): 1:00

ETE (hh:mm): [N/A]

Stability Class: D

Precipitation: None

Monitor: R14E Plnt Vnt HI

Readings: 3.40E+02 cpm

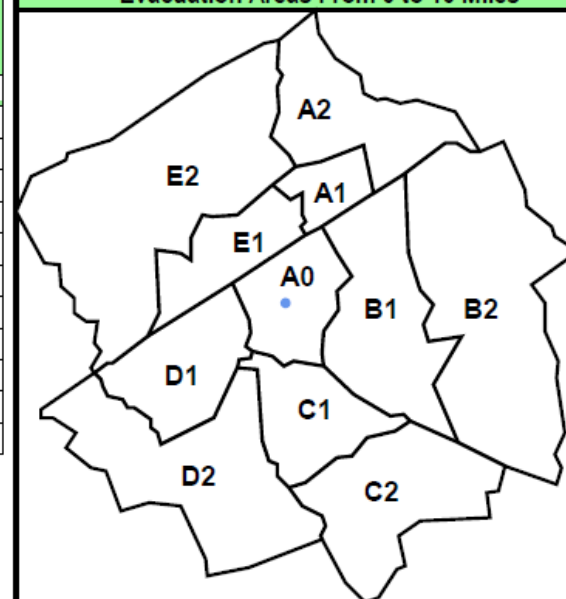
Flowrate: 60600 CFM

| Distance (Miles) | Exposure Rate (mR/hr) | External Plume DDE (mRem) | Inhalation CEDE (mRem) | Deposition Ground DDE (mRem) | TEDE (mRem) | CDE Thyroid (mRem) |
|---------------------|-----------------------------|---------------------------------|------------------------------|------------------------------------|----------------|--------------------------|
| S.B. | 1.24E+02 | 8.70E+01 | 1.00E+01 | 3.16E+00 | 1.00E+02 | 2.54E+02 |
| 0.3 | 1.13E+02 | 7.92E+01 | 8.40E+00 | 2.65E+00 | 9.02E+01 | 2.12E+02 |
| 0.5 | 7.32E+01 | 4.96E+01 | 5.68E+00 | 3.56E+00 | 5.88E+01 | 1.15E+02 |
| 0.7 | 5.24E+01 | 3.50E+01 | 4.32E+00 | 3.32E+00 | 4.26E+01 | 7.60E+01 |
| 1.0 | 3.40E+01 | 2.22E+01 | 3.04E+00 | 2.65E+00 | 2.79E+01 | 4.68E+01 |
| 1.5 | 1.84E+01 | 1.20E+01 | 1.88E+00 | 1.59E+00 | 1.54E+01 | 2.69E+01 |
| 2.0 | 1.42E+01 | 9.40E+00 | 1.29E+00 | 9.32E-01 | 1.16E+01 | 1.93E+01 |
| 3.0 | 1.19E+01 | 7.80E+00 | 8.51E-01 | 6.84E-01 | 9.34E+00 | 1.23E+01 |
| 4.0 | 1.12E+01 | 7.36E+00 | 8.54E-01 | 6.51E-01 | 8.87E+00 | 1.24E+01 |
| 5.0 | 8.68E+00 | 5.64E+00 | 7.14E-01 | 4.88E-01 | 6.85E+00 | 1.07E+01 |
| 7.0 | 5.60E+00 | 3.51E+00 | 5.03E-01 | 2.75E-01 | 4.29E+00 | 8.13E+00 |
| 10.0 | 2.77E+00 | 1.85E+00 | 3.05E-01 | 1.26E-01 | 2.28E+00 | 5.40E+00 |

Assessment Data Results Saved to File:

Robinson 10Miles Monitored Release 03092015 082947.URI7

Evacuation Areas From 0 to 10 Miles



No PAGs Exceeded

*** Classification: Site Area Emergency ***

Release Rates (Ci / sec)

| | |
|-------------|------------------|
| Particulate | 2.85E-03 (0.0%) |
| Iodine | 6.99E-02 (0.2%) |
| Noble Gas | 2.87E+01 (99.7%) |

Reviewed By: _____

Dose Assessment

Robinson

Monday, March 09, 2015 08:31

Method: Detailed Assessment - Monitored Release

Release Pathway: (I) <RCS> <Containment> <Aux Bldg> <Filters> <Plant Vent> <Env>

PRF: 1.60E-03

Containment HUT: = < 2 Hours

Cont Sprays: = OFF

Purge Filter: = N/A

Aux/Fuel Bldg HUT: = < 2 Hours

Aux/Fuel Filter: = Working

Steam Gen: = N/A

Turb Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Upper

Time After S/D (hh:mm): 0:00

Wind: From 338° @ 4.4 mph

Release Duration (hh:mm): 1:00

ETE (hh:mm): [N/A]

Stability Class: D

Precipitation: None

Monitor: R14E Plnt Vnt HI

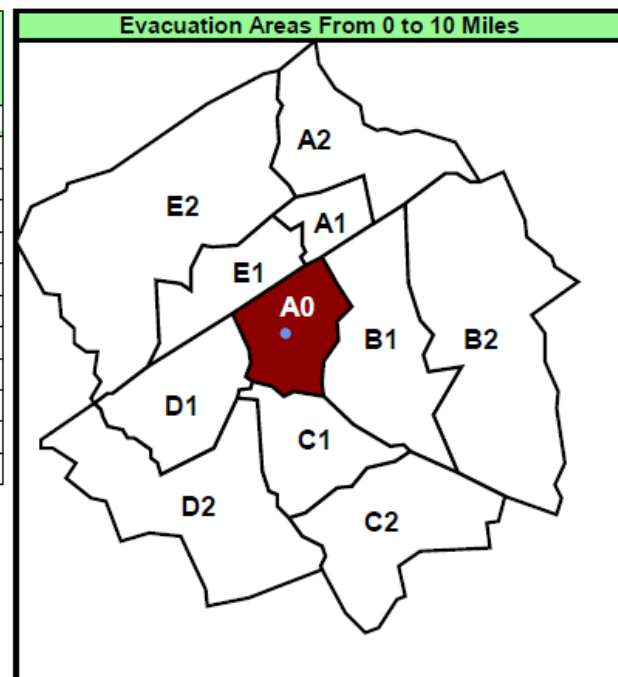
Readings: 3.31E+03 cpm

Flowrate: 60600 CFM

| Distance (Miles) | Exposure Rate (mR/hr) | External Plume DDE (mRem) | Inhalation CEDE (mRem) | Deposition Ground DDE (mRem) | TEDE (mRem) | CDE Thyroid (mRem) |
|---------------------|-----------------------------|---------------------------------|------------------------------|------------------------------------|----------------|--------------------------|
| S.B. | 1.24E+03 | 8.70E+02 | 1.00E+02 | 3.16E+01 | 1.00E+03 | 2.54E+03 |
| 0.3 | 1.13E+03 | 7.92E+02 | 8.40E+01 | 2.65E+01 | 9.02E+02 | 2.12E+03 |
| 0.5 | 7.32E+02 | 4.96E+02 | 5.68E+01 | 3.56E+01 | 5.88E+02 | 1.15E+03 |
| 0.7 | 5.24E+02 | 3.50E+02 | 4.32E+01 | 3.32E+01 | 4.26E+02 | 7.60E+02 |
| 1.0 | 3.40E+02 | 2.22E+02 | 3.04E+01 | 2.65E+01 | 2.79E+02 | 4.68E+02 |
| 1.5 | 1.84E+02 | 1.20E+02 | 1.88E+01 | 1.59E+01 | 1.54E+02 | 2.69E+02 |
| 2.0 | 1.42E+02 | 9.40E+01 | 1.29E+01 | 9.32E+00 | 1.16E+02 | 1.93E+02 |
| 3.0 | 1.19E+02 | 7.80E+01 | 8.51E+00 | 6.84E+00 | 9.34E+01 | 1.23E+02 |
| 4.0 | 1.12E+02 | 7.36E+01 | 8.54E+00 | 6.51E+00 | 8.87E+01 | 1.24E+02 |
| 5.0 | 8.68E+01 | 5.64E+01 | 7.14E+00 | 4.88E+00 | 6.85E+01 | 1.07E+02 |
| 7.0 | 5.60E+01 | 3.51E+01 | 5.03E+00 | 2.75E+00 | 4.29E+01 | 8.13E+01 |
| 10.0 | 2.77E+01 | 1.85E+01 | 3.05E+00 | 1.26E+00 | 2.28E+01 | 5.40E+01 |

Assessment Data Results Saved to File:

Robinson 10Miles Monitored Release 03092015 083154.URI7



PAGs Exceeded in Designated Areas

*** Classification: General Emergency ***

Release Rates (Ci / sec)

| | |
|-------------|------------------|
| Particulate | 2.85E-02 (0.0%) |
| Iodine | 6.99E-01 (0.2%) |
| Noble Gas | 2.87E+02 (99.7%) |

Reviewed By: _____

Dose Assessment

Robinson

Sunday, February 22, 2015 15:09

Method: Detailed Assessment - Monitored Release

Release Pathway: (T) <Waste Gas> <FHB> <Lower FHB> <Env>

PRF: 4.00E-01

Containment HUT: = N/A

Cont Sprays: = N/A

Purge Filter: = N/A

Aux/Fuel Bldg HUT: = < 2 Hours

Aux/Fuel Filter: = N/A

Steam Gen: = N/A

Turb Bldg HUT: = N/A

Source Term: Waste Gas Tank

Lower

Time After S/D (hh:mm): 0:00

Wind: From 338° @ 4.4 mph

Release Duration (hh:mm): 1:00

ETE (hh:mm): [N/A]

Stability Class: D

Precipitation: None

Monitor: R30 Lower FHB HI

Readings: 2.60E+03 mR/hr

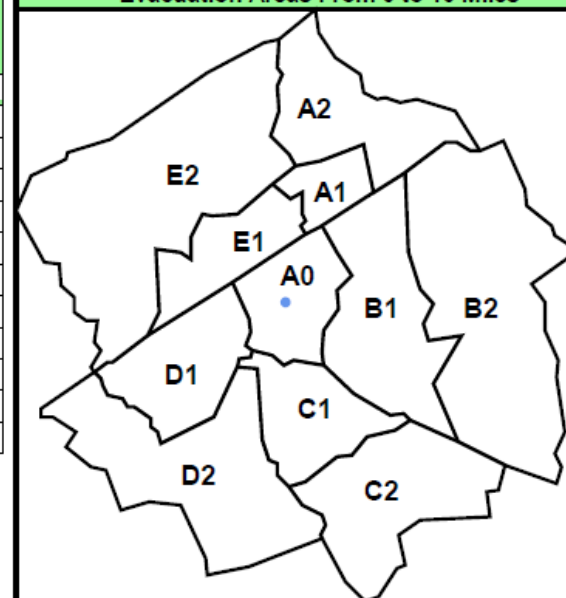
Flowrate: 10200 CFM

| Distance (Miles) | Exposure Rate (mR/hr) | External Plume DDE (mRem) | Inhalation CEDE (mRem) | Deposition Ground DDE (mRem) | TEDE (mRem) | CDE Thyroid (mRem) |
|---------------------|-----------------------------|---------------------------------|------------------------------|------------------------------------|----------------|--------------------------|
| S.B. | 1.44E+01 | 1.00E+01 | 0.00E+00 | 0.00E+00 | 1.00E+01 | 0.00E+00 |
| 0.3 | 1.20E+01 | 8.40E+00 | 0.00E+00 | 0.00E+00 | 8.40E+00 | 0.00E+00 |
| 0.5 | 6.28E+00 | 4.40E+00 | 0.00E+00 | 0.00E+00 | 4.40E+00 | 0.00E+00 |
| 0.7 | 3.90E+00 | 2.73E+00 | 0.00E+00 | 0.00E+00 | 2.73E+00 | 0.00E+00 |
| 1.0 | 2.24E+00 | 1.57E+00 | 0.00E+00 | 0.00E+00 | 1.57E+00 | 0.00E+00 |
| 1.5 | 1.56E+00 | 1.09E+00 | 0.00E+00 | 0.00E+00 | 1.09E+00 | 0.00E+00 |
| 2.0 | 5.60E-01 | 3.91E-01 | 0.00E+00 | 0.00E+00 | 3.91E-01 | 0.00E+00 |
| 3.0 | 7.72E-01 | 5.40E-01 | 0.00E+00 | 0.00E+00 | 5.40E-01 | 0.00E+00 |
| 4.0 | 8.16E-01 | 5.69E-01 | 0.00E+00 | 0.00E+00 | 5.69E-01 | 0.00E+00 |
| 5.0 | 7.12E-01 | 4.94E-01 | 0.00E+00 | 0.00E+00 | 4.94E-01 | 0.00E+00 |
| 7.0 | 5.68E-01 | 3.78E-01 | 0.00E+00 | 0.00E+00 | 3.78E-01 | 0.00E+00 |
| 10.0 | 3.73E-01 | 2.63E-01 | 0.00E+00 | 0.00E+00 | 2.63E-01 | 0.00E+00 |

Assessment Data Results Saved to File:

Robinson 10Miles Monitored Release 02222015 150931.URI7

Evacuation Areas From 0 to 10 Miles



No PAGs Exceeded

Release Rates (Ci / sec)

| | |
|-------------|-------------------|
| Particulate | 0.00E+00 (0.0%) |
| Iodine | 0.00E+00 (0.0%) |
| Noble Gas | 5.01E+01 (100.0%) |

*** Classification: Validate against Emergency Action Levels ***

Reviewed By: _____

Dose Assessment

Robinson

Sunday, February 22, 2015 15:24

Method: Detailed Assessment - Monitored Release

Release Pathway: (T) <Waste Gas> <FHB> <Lower FHB> <Env>

PRF: 4.00E-01

Containment HUT: = N/A

Cont Sprays: = N/A

Purge Filter: = N/A

Aux/Fuel Bldg HUT: = < 2 Hours

Aux/Fuel Filter: = N/A

Steam Gen: = N/A

Turb Bldg HUT: = N/A

Source Term: Waste Gas Tank

Lower

Time After S/D (hh:mm): 0:00

Wind: From 338° @ 4.4 mph

Release Duration (hh:mm): 1:00

ETE (hh:mm): [N/A]

Stability Class: D

Precipitation: None

Monitor: R30 Lower FHB HI

Readings: 2.60E+04 mR/hr

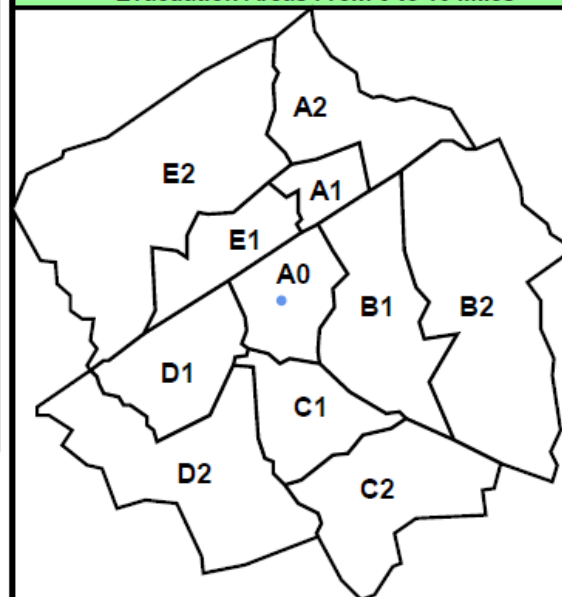
Flowrate: 10200 CFM

| Distance (Miles) | Exposure Rate (mR/hr) | External Plume DDE (mRem) | Inhalation CEDE (mRem) | Deposition Ground DDE (mRem) | TEDE (mRem) | CDE Thyroid (mRem) |
|---------------------|-----------------------------|---------------------------------|------------------------------|------------------------------------|----------------|--------------------------|
| S.B. | 1.44E+02 | 1.00E+02 | 0.00E+00 | 0.00E+00 | 1.00E+02 | 0.00E+00 |
| 0.3 | 1.20E+02 | 8.40E+01 | 0.00E+00 | 0.00E+00 | 8.40E+01 | 0.00E+00 |
| 0.5 | 6.28E+01 | 4.40E+01 | 0.00E+00 | 0.00E+00 | 4.40E+01 | 0.00E+00 |
| 0.7 | 3.90E+01 | 2.73E+01 | 0.00E+00 | 0.00E+00 | 2.73E+01 | 0.00E+00 |
| 1.0 | 2.24E+01 | 1.57E+01 | 0.00E+00 | 0.00E+00 | 1.57E+01 | 0.00E+00 |
| 1.5 | 1.56E+01 | 1.09E+01 | 0.00E+00 | 0.00E+00 | 1.09E+01 | 0.00E+00 |
| 2.0 | 5.60E+00 | 3.91E+00 | 0.00E+00 | 0.00E+00 | 3.91E+00 | 0.00E+00 |
| 3.0 | 7.72E+00 | 5.40E+00 | 0.00E+00 | 0.00E+00 | 5.40E+00 | 0.00E+00 |
| 4.0 | 8.16E+00 | 5.69E+00 | 0.00E+00 | 0.00E+00 | 5.69E+00 | 0.00E+00 |
| 5.0 | 7.12E+00 | 4.94E+00 | 0.00E+00 | 0.00E+00 | 4.94E+00 | 0.00E+00 |
| 7.0 | 5.68E+00 | 3.78E+00 | 0.00E+00 | 0.00E+00 | 3.78E+00 | 0.00E+00 |
| 10.0 | 3.73E+00 | 2.63E+00 | 0.00E+00 | 0.00E+00 | 2.63E+00 | 0.00E+00 |

Assessment Data Results Saved to File:

Robinson 10Miles Monitored Release 02222015 152408.URI7

Evacuation Areas From 0 to 10 Miles



No PAGs Exceeded

Release Rates (Ci / sec)

| | |
|-------------|-------------------|
| Particulate | 0.00E+00 (0.0%) |
| Iodine | 0.00E+00 (0.0%) |
| Noble Gas | 5.01E+02 (100.0%) |

*** Classification: Site Area Emergency ***

Reviewed By: _____

RNP-RA/15-0034
Enclosure 6
3 Pages (including cover page)

Enclosure 6

EMERGENCY ACTION LEVEL WALLCHARTS FOR HBRSEP2

| | | GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
|---------|-----------------------------------|---|---|---|---|----------------------------|-----|-------|----|---------|------------|-------------------------|----------------------------|----------------------------|----------------------------|----------------------------|-------------|------|-----|-----|-----|-------------|----------------|------|-----|---------------|---------------|-----|-----------------------|------|-----|-----|-----|-------------|--------|---------------|------------|-----|-----|-----|-------------|---------------------|------|-----|-----|-----|-------------|------|------|------|------|------|------|------|--|
| R | Abnorm. Rad Levels / Rad Effluent | <div>Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE</div> <div>123456DEF</div> <div>RG1.1 Reading on any Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4)</div> <div>RG1.2 Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 3, 4)</div> <div>RG1.3 Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY: - Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min. - Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation. (Notes 1, 2)</div> | <div>Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE</div> <div>123456DEF</div> <div>RS1.1 Reading on any Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4)</div> <div>RS1.2 Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 3, 4)</div> <div>RS1.3 Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY: - Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min. - Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation. (Notes 1, 2)</div> | <div>Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE</div> <div>123456DEF</div> <div>RA1.1 Reading on any Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4)</div> <div>RA1.2 Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 3, 4)</div> <div>RA1.3 Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure (Notes 1, 2)</div> <div>RA1.4 Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY: - Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min. - Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation. (Notes 1, 2)</div> | <div>Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer</div> <div>123456DEF</div> <div>RU1.1 Reading on any Table R-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3)</div> <div>RU1.2 Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2)</div> | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| | | <div>Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer</div> <div>123456DEF</div> <div>RG2.1 Spent fuel pool level cannot be restored to at least 14.75 ft. for ≥ 60 min. (Note 1)</div> | <div>Spent fuel pool level at the top of the fuel racks</div> <div>123456DEF</div> <div>RS2.1 Lowering of spent fuel pool level to ≤ 14.75 ft.</div> | <div>Significant lowering of water level above, or damage to, irradiated fuel</div> <div>123456DEF</div> <div>RA2.1 Uncovery of irradiated fuel in the REFUELING PATHWAY</div> <div>RA2.2 Damage to irradiated fuel resulting in a release of radioactivity AND A high alarm on any of the following: - R-2 CV Area - R-5 Spent Fuel Pit Area - R-11/R-12 Process Monitor CV Air and Plant Vent (when sampling CV) - R-14 Plant Vent - R-21 Fuel Handling Building Upper Level</div> <div>RA2.3 Lowering of spent fuel pool level to ≤ 24 ft.</div> | <div>Unplanned loss of water level above irradiated fuel</div> <div>123456DEF</div> <div>RU2.1 UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication AND UNPLANNED rise in corresponding area radiation levels as indicated by any of the following: - R-2 CV Area - R-5 Spent Fuel Pit Area - Local area survey</div> | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| | | <div>Table R-1 Effluent Monitor Classification Thresholds</div> <table><thead><tr><th></th><th>Release Point</th><th>Monitor</th><th>GE</th><th>SAE</th><th>Alert</th><th>UE</th></tr></thead><tbody><tr><td rowspan="4">Gaseous</td><td>Plant Vent</td><td>R-14C R-14D R-14E</td><td>6.38E+5 cpm 3.31E+3 cpm</td><td>6.38E+4 cpm 3.40E+2 cpm</td><td>6.38E+3 cpm 4.30E+1 cpm</td><td>2.16E+5 cpm 4.06E+6 cpm</td></tr><tr><td>FHB Exhaust</td><td>R-20</td><td>---</td><td>---</td><td>---</td><td>8.06E+5 cpm</td></tr><tr><td>FHB Exhaust HR</td><td>R-30</td><td>---</td><td>2.60E+4 mR/hr</td><td>2.60E+3 mR/hr</td><td>---</td></tr><tr><td>Liquid Waste Disposal</td><td>R-18</td><td>---</td><td>---</td><td>---</td><td>4.08E+4 cpm</td></tr><tr><td rowspan="2">Liquid</td><td>SGBD Effluent</td><td>R-19 A/B/C</td><td>---</td><td>---</td><td>---</td><td>6.94E+5 cpm</td></tr><tr><td>Condensate Polisher</td><td>R-37</td><td>---</td><td>---</td><td>---</td><td>4.23E+5 cpm</td></tr><tr><td>None</td><td>None</td><td>None</td><td>None</td><td>None</td><td>None</td><td>None</td></tr></tbody></table> | | Release Point | Monitor | GE | SAE | Alert | UE | Gaseous | Plant Vent | R-14C R-14D R-14E | 6.38E+5 cpm 3.31E+3 cpm | 6.38E+4 cpm 3.40E+2 cpm | 6.38E+3 cpm 4.30E+1 cpm | 2.16E+5 cpm 4.06E+6 cpm | FHB Exhaust | R-20 | --- | --- | --- | 8.06E+5 cpm | FHB Exhaust HR | R-30 | --- | 2.60E+4 mR/hr | 2.60E+3 mR/hr | --- | Liquid Waste Disposal | R-18 | --- | --- | --- | 4.08E+4 cpm | Liquid | SGBD Effluent | R-19 A/B/C | --- | --- | --- | 6.94E+5 cpm | Condensate Polisher | R-37 | --- | --- | --- | 4.23E+5 cpm | None | None | None | None | None | None | None | |
| | Release Point | Monitor | GE | SAE | Alert | UE | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| Gaseous | Plant Vent | R-14C R-14D R-14E | 6.38E+5 cpm 3.31E+3 cpm | 6.38E+4 cpm 3.40E+2 cpm | 6.38E+3 cpm 4.30E+1 cpm | 2.16E+5 cpm 4.06E+6 cpm | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| | FHB Exhaust | R-20 | --- | --- | --- | 8.06E+5 cpm | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| | FHB Exhaust HR | R-30 | --- | 2.60E+4 mR/hr | 2.60E+3 mR/hr | --- | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| | Liquid Waste Disposal | R-18 | --- | --- | --- | 4.08E+4 cpm | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| Liquid | SGBD Effluent | R-19 A/B/C | --- | --- | --- | 6.94E+5 cpm | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| | Condensate Polisher | R-37 | --- | --- | --- | 4.23E+5 cpm | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
| None | None | None | None | None | None | None | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |

| H | Security | Hostile Action resulting in loss of physical control of the facility 123456DEF HG1.1 A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision **AND EITHER** of the following has occurred: Any of the following safety functions cannot be controlled or maintained - Reactivity - Core Cooling - RCS heat removal **OR** Damage to spent fuel has occurred or is IMMINENT | Hostile Action within the Protected Area 123456DEF HS1.1 A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision | Hostile Action within the Owner Controlled Area or airborne attack threat within 30 minutes 123456DEF HA1.1 A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervision **OR** A validated notification from NRC of an aircraft attack threat within 30 min. of the site | Confirmed SECURITY CONDITION or threat 123456DEF HU1.1 A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the 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 |
| Table R-2/H-2 Safe Operation & Shutdown Rooms/Areas | Room / Area | Mode(s) | |---|---------------| | - Reactor Auxiliary Building, 1st level hallway | 1, 2, 3, 4, 5 | | - Reactor Auxiliary Building, 2nd level hallway | 1, 2, 3, 4, 5 | | - Charging Pump Room | 1, 2, 3, 4, 5 | | - Component Cooling Water Pump Room | 1, 2, 3, 4, 5 | | - Primary Sample Room | 1, 2, 3, 4, 5 | | - Primary Demineralizer Room | 1, 2, 3 | | - Spent Fuel Pump / Heat Exchanger Room | 1, 2, 3, 4, 5 | | - Pipe Alley | 4 | | - RHR Heat Exchanger Room | 4 | | - RHR Pump Room entry area (access to RHR Pump CCW flow indication / control) | 4 | | - Boric Acid Batch Tank Room | 1, 2, 3, 4, 5 | | - Emergency Bus E1/E2 Room | 3, 4, 5 | | - Turbine Building 1st Floor (includes Condensate Polisher, Makeup Water Treatment and Secondary Sample Room) | 1, 2, 3, 4 | | - Turbine Building 2nd Floor | 1, 2, 3, 4 | | - Turbine Building 3rd Floor | 1, 3, 4 | | - Containment Building | 3 | | | [Refer to EAL HA6.1 OR SA9.1 for escalation due to seismic event] None | Seismic event greater than OBE levels 123456DEF HU2.1 Seismic Recording Unit A or B indicates seismic event > Operating Basis Earthquake (0.1g horizontal **OR** 0.067g vertical) |
| Table R-2/H-2 Safe Operation & Shutdown Rooms/Areas | Room / Area | Mode(s) | |---|---------------| | - Reactor Auxiliary Building, 1st level hallway | 1, 2, 3, 4, 5 | | - Reactor Auxiliary Building, 2nd level hallway | 1, 2, 3, 4, 5 | | - Charging Pump Room | 1, 2, 3, 4, 5 | | - Component Cooling Water Pump Room | 1, 2, 3, 4, 5 | | - Primary Sample Room | 1, 2, 3, 4, 5 | | - Primary Demineralizer Room | 1, 2, 3 | | - Spent Fuel Pump / Heat Exchanger Room | 1, 2, 3, 4, 5 | | - Pipe Alley | 4 | | - RHR Heat Exchanger Room | 4 | | - RHR Pump Room entry area (access to RHR Pump CCW flow indication / control) | 4 | | - Boric Acid Batch Tank Room | 1, 2, 3, 4, 5 | | - Emergency Bus E1/E2 Room | 3, 4, 5 | | - Turbine Building 1st Floor (includes Condensate Polisher, Makeup Water Treatment and Secondary Sample Room) | 1, 2, 3, 4 | | - Turbine Building 2nd Floor | 1, 2, 3, 4 | | - Turbine Building 3rd Floor | 1, 3, 4 | | - Containment Building | 3 | | | [Refer to EAL HA6.1 OR SA9.1 for escalation due to natural or technological hazard] None | FIRE potentially degrading the level of safety of the plant 123456DEF HU4.1 A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1): - Report from the field (i.e., visual observation) - Receipt of multiple (more than 1) fire alarms or indications - Field verification of a single fire alarm **AND** The FIRE is located within any Table H-1 area **HU4.2** Receipt of a single fire alarm (i.e., no other indications of a FIRE) **AND** The fire alarm is indicating a FIRE within any Table H-1 area **HU4.3** A FIRE within the plant PROTECTED AREA not extinguished within 60 min. of the initial report, alarm or indication (Note 1) **HU4.4** A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish |
| E | ISFSI | Other conditions exist which in the judgment of the Site Emergency Coordinator warrant declaration of a General Emergency 123456DEF HG7.1 Other conditions exist which in the judgment of the Emergency Coordinator 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 | Other conditions existing that in the judgment of the Site Emergency Coordinator warrant declaration of a Site Area Emergency 123456DEF HS7.1 Other conditions exist which in the judgment of the Emergency Coordinator 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. | Other conditions exist that in the judgment of the Site Emergency Coordinator warrant declaration of an Alert 123456DEF HA7.1 Other conditions exist which, in the judgment of the Emergency Coordinator, 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. | Other conditions existing that in the judgment of the Site Emergency Coordinator warrant declaration of a UE 123456DEF HU7.1 Other conditions exist which in the judgment of the Emergency Coordinator 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 | Table E-1 ISFSI Dose Limits | 7P ISFSI | 24P ISFSI | |--|--| | • 400 mrem/hr outside of HSM door on centerline of DSC | • 2,600 mrem/hr on the HSM front surface | | • 400 mrem/hr at center of air inlets or outlets | • 10 mrem/hr on the HSM-H door centerline | | • 100 mrem/hr on roof, front, back or side | • 20 mrem/hr on the end shield wall exterior | | | Damage to a loaded cask CONFINEMENT BOUNDARY 123456DEF EU1.1 Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask > **any** Table E-1 dose limit |
| Modes: | | 1 Power Operations 2 Startup 3 Hot Standby 4 Hot Shutdown 5 Cold Shutdown 6 Refuel DEF Defueled | DUKE ENERGY. Robinson Nuclear Plant Classification of Emergency EPLCA-04 Rev. 0 |

| | | GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT | | | | | | | | | | | | |
|---|--|--|--|---|---|------------------|------------------------------------|-----|----------|--|-------------|----------|--|------------------------|--------|---|---|
| C | 1 RCS Level | <p>Loss of RCS inventory affecting fuel clad integrity with containment challenged</p> <div><div></div><div></div><div></div><div></div><div>5</div><div>6</div></div> <p>CG1.1 RCS level < 59.8% RVLIS Full Range for ≥ 30 min. (Note 1) AND Any Containment Challenge indication, Table C-2</p> <p>CG1.2 RCS water level cannot be monitored for ≥ 30 min. (Note 1) AND Core uncovery is indicated by any of the following:</p> <ul style="list-style-type: none">- UNPLANNED increase in any Table C-1 sump or tank due to a loss of RCS inventory- Visual observation of UNISOLABLE RCS leakage- Containment High Range Radiation Monitor R-32A or R-32B > 5 Rem/hr- Erratic source range monitor indication <p>AND Any Containment Challenge indication, Table C-2</p> | <p>Loss of RCS inventory affecting core decay heat removal capability</p> <div><div></div><div></div><div></div><div></div><div>5</div><div>6</div></div> <p>CS1.1 With CONTAINMENT CLOSURE not established, RCS level < 64.5% RVLIS Full Range</p> <p>CS1.2 With CONTAINMENT CLOSURE established, RCS level < 59.8% RVLIS Full Range</p> <p>CS1.3 RCS water level cannot be monitored for ≥ 30 min. (Note 1) AND Core uncovery is indicated by any of the following:</p> <ul style="list-style-type: none">- UNPLANNED increase in any Table C-1 sump or tank due to a loss of RCS inventory- Visual observation of UNISOLABLE RCS leakage- Containment High Range Radiation Monitor R-32A or R-32B > 5 Rem/hr- Erratic source range monitor indication | <p>Loss of RCS inventory</p> <div><div></div><div></div><div></div><div></div><div>5</div><div>6</div></div> <p>CA1.1 Loss of RCS inventory as indicated by RCS water level < 72 in. (69% RVLIS Full Range)</p> <p>CA1.2 RCS water level cannot be monitored for ≥ 15 min. (Note 1) AND EITHER</p> <ul style="list-style-type: none">- UNPLANNED increase in any Table C-1 sump or tank due to a loss of RCS inventory- Visual observation of UNISOLABLE RCS leakage | <p>UNPLANNED loss of RCS inventory for 15 minutes or longer</p> <div><div></div><div></div><div></div><div></div><div>5</div><div>6</div></div> <p>CU1.1 UNPLANNED loss of reactor coolant results in RCS water level less than a required lower limit for ≥ 15 min. (Note 1)</p> <p>CU1.2 RCS water level cannot be monitored AND EITHER</p> <ul style="list-style-type: none">- UNPLANNED increase in any Table C-1 sump or tank due to a loss of RCS inventory- Visual observation of UNISOLABLE RCS leakage | | | | | | | | | | | | |
| | 2 Loss of Emergency AC Power | <div><div>Table C-2 Containment Challenge Indications</div><div><ul style="list-style-type: none">• Containment Closure not established (Note 6)• Containment hydrogen concentration ≥ 4%• UNPLANNED rise in Containment pressure</div></div> | <div><div>Table C-1 Sumps / Tanks</div><div><ul style="list-style-type: none">• Containment (CV) sump• PRT• RCDT• CCW surge tank</div></div> | <p>Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer</p> <div><div></div><div></div><div></div><div></div><div>5</div><div>6</div><div>DEF</div></div> <p>CA2.1 Loss of all offsite and all onsite AC power capability to 480V emergency buses E-1 and E-2 for ≥ 15 min. (Note 1)</p> | <p>Loss of all but one AC power source to emergency buses for 15 minutes or longer</p> <div><div></div><div></div><div></div><div></div><div>5</div><div>6</div><div>DEF</div></div> <p>CU2.1 AC power capability to 480V emergency buses E-1 and E-2 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</p> | | | | | | | | | | | | |
| | 3 RCS Temp. | None | <div><div>Table C-3 RCS Re-heat Duration Thresholds</div><table><tr><th>RCS Status</th><th>Containment Closure Status</th><th>Heat-up Duration</th></tr><tr><td>Intact (but not REDUCED INVENTORY)</td><td>N/A</td><td>60 min.*</td></tr><tr><td>Not intact OR At REDUCED INVENTORY</td><td>established</td><td>20 min.*</td></tr><tr><td></td><td>not established</td><td>0 min.</td></tr></table><p>* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable</p></div> | RCS Status | Containment Closure Status | Heat-up Duration | Intact (but not REDUCED INVENTORY) | N/A | 60 min.* | Not intact OR At REDUCED INVENTORY | established | 20 min.* | | not established | 0 min. | <p>Inability to maintain plant in cold shutdown</p> <div><div></div><div></div><div></div><div></div><div>5</div><div>6</div></div> <p>CA3.1 UNPLANNED increase in RCS temperature to > 200°F for > Table C-3 duration (Note 1) OR UNPLANNED RCS pressure increase > 10 psig due to a loss of RCS cooling cooling (this does not apply during water-solid plant conditions)</p> | <p>UNPLANNED increase in RCS temperature</p> <div><div></div><div></div><div></div><div></div><div>5</div><div>6</div></div> <p>CU3.1 UNPLANNED increase in RCS temperature to > 200°F due to loss of decay heat removal capability</p> <p>CU3.2 Loss of all RCS temperature and RCS level indication for ≥ 15 min. (Note 1)</p> |
| | RCS Status | Containment Closure Status | Heat-up Duration | | | | | | | | | | | | | | |
| | Intact (but not REDUCED INVENTORY) | N/A | 60 min.* | | | | | | | | | | | | | | |
| | Not intact OR At REDUCED INVENTORY | established | 20 min.* | | | | | | | | | | | | | | |
| | not established | 0 min. | | | | | | | | | | | | | | | |
| 4 Loss of Vital DC Power | None | | None | <p>Loss of Vital DC power for 15 minutes or longer</p> <div><div></div><div></div><div></div><div></div><div>5</div><div>6</div></div> <p>CU4.1 < 109.5 VDC (Bus A) / < 106.2 (Bus B) bus voltage indications on Technical Specification required 125 VDC buses for ≥ 15 min. (Note 1)</p> | | | | | | | | | | | | | |
| 5 Loss of Comm. | None | None | None | <p>Loss of all onsite or offsite communications capabilities</p> <div><div></div><div></div><div></div><div></div><div>5</div><div>6</div><div>DEF</div></div> <p>CU5.1 Loss of all Table C-4 onsite communication methods OR Loss of all Table C-4 offsite communication methods OR Loss of all Table C-4 NRC communication methods</p> | | | | | | | | | | | | | |
| 6 Hazardous Event Affecting Safety Systems | None | <div><div>Table C-5 Hazardous Events</div><div><ul style="list-style-type: none">- 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</div></div> | <p>Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode</p> <div><div></div><div></div><div></div><div></div><div>5</div><div>6</div></div> <p>CA6.1 The occurrence of any Table C-5 hazardous event AND EITHER:</p> <ul style="list-style-type: none">- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode | None | | | | | | | | | | | | | |

| Reactor Vessel Levels | | | | |
|--------------------------|------------------|-----------|-------------|--|
| | RVLIS Full Range | Standpipe | EAL | |
| Reactor Vessel Flange | --- | 0" | --- | |
| Min. RHR Operation | 69.0% | -72" | CA1.1 | |
| Bottom of Hotleg | 65.8% | --- | CS1.1 | |
| 6 in. < Bottom of Hotleg | 64.5% | --- | CS1.1/CG1.1 | |
| Top of Active Fuel | 59.8% | --- | CS1.1/CG1.1 | |

| NOTES | |
|--|--|
| Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded. Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit. Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes. Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available. Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted. Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is not required. Note 7: This EAL does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents. Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies. Note 9: Mode 3 applicable only when RCS temperature is ≥ 500°F | |

| Table C-4 Communication Methods | | | |
|--|--------|---------|-----|
| System | Onsite | Offsite | NRC |
| Public Address System | X | | |
| PBX Telephone System | X | | |
| Radio Transceivers for RNP and Vicinity | X | | |
| Back-up Telephone System (ESSX) | X | | |
| Plant Security Radio Transceivers | X | | |
| Corporate Telephone Communications System (Voiceint) | | X | X |
| BellSouth | | X | X |
| Dedicated Telephone System to Load Dispatcher | | X | |
| Plant Security Radio Control Station | | X | |
| DEMNET | | X | |
| NRC Emergency Telecommunication System (ETS) | | | X |
| Satellite Phones | | X | X |
| Cellular Phones | | X | X |
| Palmetto 800 Transceivers | | X | |

EAL - COLD MODES 5, 6 & Defueled

| | | GENERAL EMERGENCY | | SITE AREA EMERGENCY | | ALERT | | UNUSUAL EVENT | |
|---|-------------------------|--|--|--|--|--|---|---------------|--|
| R | 1 | Rad Effluent | Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE 1 2 3 4 5 6 DEF RG1.1 Reading on any Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4) RG1.2 Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 3, 4) RG1.3 Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY: - Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min. - Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation. (Notes 1, 2) | Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE 1 2 3 4 5 6 DEF RS1.1 Reading on any Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4) RS1.2 Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 3, 4) RS1.3 Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY: - Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min. - Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation. (Notes 1, 2) | Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE 1 2 3 4 5 6 DEF RA1.1 Reading on any Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4) RA1.2 Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 3, 4) RA1.3 Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure (Notes 1, 2) RA1.4 Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY: - Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min. - Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation. (Notes 1, 2) | Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer 1 2 3 4 5 6 DEF RU1.1 Reading on any Table R-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3) RU1.2 Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2) | | | |
| | | | Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer. 1 2 3 4 5 6 DEF RG2.1 Spent fuel pool level cannot be restored to at least 14.75 ft. for ≥ 60 min. (Note 1) | Spent fuel pool level at the top of the fuel racks 1 2 3 4 5 6 DEF RS2.1 Lowering of spent fuel pool level to ≤ 14.75 ft. | Significant lowering of water level above, or damage to, irradiated fuel 1 2 3 4 5 6 DEF RA2.1 Uncovery of irradiated fuel in the REFUELING PATHWAY RA2.2 Damage to irradiated fuel resulting in a release of radioactivity AND A high alarm on any of the following: - R-2 CV Area - R-5 Spent Fuel Pit Area - R-11/R-12 Process Monitor CV Air and Plant Vent (when sampling CV) - R-14 Plant Vent - R-21 Fuel Handling Building Upper Level RA2.3 Lowering of spent fuel pool level to ≤ 24 ft. Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown 1 2 3 4 5 6 DEF RA3.1 Dose rates > 15 mR/hr in EITHER of the following areas: Control Room (R-1) OR Central Alarm Station (by survey) RA3.2 An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to any Table R-2/H-2 rooms or areas (Note 5) | Unplanned loss of water level above irradiated fuel 1 2 3 4 5 6 DEF RU2.1 UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication AND UNPLANNED rise in corresponding area radiation levels as indicated by any of the following: - R-2 CV Area - R-5 Spent Fuel Pit Area - Local area survey | | | |
| | 2 | Irradiated Fuel Event | Table R-1 Effluent Monitor Classification Thresholds | | Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown 1 2 3 4 5 6 DEF RA3.1 Dose rates > 15 mR/hr in EITHER of the following areas: Control Room (R-1) OR Central Alarm Station (by survey) RA3.2 An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to any Table R-2/H-2 rooms or areas (Note 5) | | | | |
| | | | None | | None | | None | | |
| H | 1 | Security | Hostile Action resulting in loss of physical control of the facility 1 2 3 4 5 6 DEF HG1.1 A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision AND EITHER of the following has occurred: Any of the following safety functions cannot be controlled or maintained - Reactivity - Core Cooling - RCS heat removal OR Damage to spent fuel has occurred or is IMMINENT | Hostile Action within the Protected Area 1 2 3 4 5 6 DEF HS1.1 A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision | Hostile Action within the Owner Controlled Area or airborne attack threat within 30 minutes 1 2 3 4 5 6 DEF HA1.1 A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervision OR A validated notification from NRC of an aircraft attack threat within 30 min. of the site | Confirmed SECURITY CONDITION or threat 1 2 3 4 5 6 DEF HU1.1 A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the 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 | | | |
| | | | Table R-2/H-2 Safe Operation & Shutdown Rooms/Areas | | None [Refer to EAL HA6.1 OR SA8.1 for escalation due to seismic event] | | Seismic event greater than OBE levels 1 2 3 4 5 6 DEF HU2.1 Seismic Recording Unit A or B indicates seismic event > Operating Basis Earthquake (0.1g horizontal OR 0.067g vertical) | | |
| | 2 | Seismic Event | Table R-2/H-2 Safe Operation & Shutdown Rooms/Areas | | None [Refer to EAL HA6.1 OR SA8.1 for escalation due to natural or technological hazard] | | Hazardous event 1 2 3 4 5 6 DEF HU3.1 A tornado strike within the PROTECTED AREA HU3.2 Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode HU3.3 Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release) HU3.4 A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7) | | |
| | | | None | | Table H-1 Fire Areas | | FIRE potentially degrading the level of safety of the plant 1 2 3 4 5 6 DEF HU4.1 A FIRE is not extinguished within 15 min. of any of the following FIRE detection indications (Note 1): - Report from the field (i.e., visual observation) - Receipt of multiple (more than 1) fire alarms or indications - Field verification of a single fire alarm AND The FIRE is located within any Table H-1 area HU4.2 Receipt of a single fire alarm (i.e., no other indications of a FIRE) AND The fire alarm is indicating a FIRE within any Table H-1 area AND The existence of a FIRE is not verified within 30 min. of alarm receipt (Note 1) HU4.3 A FIRE within the plant PROTECTED AREA not extinguished within 60 min. of the initial report, alarm or indication (Note 1) HU4.4 A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish | | |
| 3 | Natural or Tech. Hazard | NOTES | | None | | | | | |
| | | None | | None | | | | | |
| E | 4 | Fire | NOTES | | None | | | | |
| | | | None | | None | | | | |
| | 5 | Hazardous Gases | None | None | Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown 1 2 3 4 5 6 DEF HA5.1 Release of a toxic, corrosive, asphyxiant or flammable gas into any Table R-2/H-2 rooms or areas AND Entry into the room or area is prohibited or IMPEDED (Note 5) | | None | | |
| | | | None | Inability to control a key safety function from outside the Control Room 1 2 3 4 5 6 DEF HS6.1 An event has resulted in plant control being transferred from the Control Room to the Dedicated/Alternate Shutdown System AND Control of any of the following key safety functions is not reestablished within 15 min. (Note 1): - Reactivity - Core Cooling - RCS heat removal | Control Room evacuation resulting in transfer of plant control to alternate locations 1 2 3 4 5 6 DEF HA6.1 An event has resulted in plant control being transferred from the Control Room to the Dedicated/Alternate Shutdown System | None | | | |
| 6 | Control Room Evacuation | None | None | None | | None | | | |
| | | Other conditions exist which in the judgment of the Site Emergency Coordinator warrant declaration of a General Emergency 1 2 3 4 5 6 DEF HG7.1 Other conditions exist which in the judgment of the Emergency Coordinator 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 | Other conditions existing that in the judgment of the Site Emergency Coordinator warrant declaration of a Site Area Emergency 1 2 3 4 5 6 DEF HS7.1 Other conditions exist which in the judgment of the Emergency Coordinator 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. | Other conditions exist that in the judgment of the Site Emergency Coordinator warrant declaration of an Alert 1 2 3 4 5 6 DEF HA7.1 Other conditions exist which, in the judgment of the Emergency Coordinator, indicate that events are in progress or have occurred which indicate a 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. | Other conditions existing that in the judgment of the Site Emergency Coordinator warrant declaration of a UE 1 2 3 4 5 6 DEF HU7.1 Other conditions exist which in the judgment of the Emergency Coordinator 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. | | | | |
| 7 | EC Judgment | None | | Table E-1 ISFSI Dose Limits | | Damage to a loaded cask CONFINEMENT BOUNDARY 1 2 3 4 5 6 DEF EU1.1 Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask > any Table E-1 dose limit | | | |
| | | None | | 7P ISFSI • 400 mrem/hr outside of HSM door on centerline of DSC • 400 mrem/hr at center of air inlets or outlets • 100 mrem/hr on roof, front, back or side 24P ISFSI • 2,600 mrem/hr on the HSM front surface • 10 mrem/hr on the HSM-H door centerline • 20 mrem/hr on the end shield wall exterior | | | | | |

Robinson Nuclear Plant Classification of Emergency EPLA-04 Rev. 0

Modes:

1

2

3

4

5

6

DEF

Defueled



Robinson Nuclear Plant
Classification of Emergency
EPLA-04 Rev. 0

| | | GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT |
|---|---|---|---|--|---|
| S System Malfunc. | 1 Loss of Emergency AC Power | <p>Prolonged loss of all offsite and all onsite AC power to emergency buses</p> <p>1 2 3 4</p> <p>SG1.1 Loss of all offsite and all onsite AC power capability to 480V emergency buses E-1 and E-2 AND EITHER</p> <ul style="list-style-type: none">- Restoration of at least one emergency bus in < 8 hours is not likely (Note 1)- Core Cooling RED Path entry conditions met <p>Loss of all AC and vital DC power sources for 15 minutes or longer</p> <p>1 2 3 4</p> <p>SG1.2 Loss of all offsite and all onsite AC power capability to 480V emergency buses E-1 and E-2 for ≥ 15 min. AND Loss of all vital DC power based on < 109.5 VDC Bus A and < 108.2 VDC Bus B voltage indications for ≥ 15 min. (Note 1)</p> | <p>Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer</p> <p>1 2 3 4</p> <p>SS1.1 Loss of all offsite and all onsite AC power capability to 480V emergency buses E-1 and E-2 for ≥ 15 min. (Note 1)</p> <p>Loss of all vital DC power for 15 minutes or longer</p> <p>1 2 3 4</p> <p>SS2.1 Loss of all vital DC power based on < 109.5 VDC Bus A and < 108.2 VDC Bus B voltage indications for ≥ 15 min. (Note 1)</p> | <p>Loss of all but one AC power source to emergency buses for 15 minutes or longer</p> <p>1 2 3 4</p> <p>SA1.1 AC power capability to 480V emergency buses E-1 and E-2 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</p> | <p>Loss of all offsite AC power capability to emergency buses for 15 minutes or longer</p> <p>1 2 3 4</p> <p>SU1.1 Loss of all offsite AC power capability to 480V emergency buses E-1 and E-2 for ≥ 15 min. (Note 1)</p> |
| | 2 Loss of Vital DC Power | | | None | None |
| | 3 Loss of CR Indications | None | <p>Table S-2 Significant Transients</p> <ul style="list-style-type: none">- Reactor trip- Runback > 25% thermal power- Electrical load rejection > 25% electrical load- Safety injection action | <p>UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress</p> <p>1 2 3 4</p> <p>SA3.1 An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1) AND Any significant transient is in progress, Table S-2</p> | <p>UNPLANNED loss of Control Room indications for 15 minutes or longer</p> <p>1 2 3 4</p> <p>SU3.1 An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1)</p> |
| | 4 RCS Activity | None | None | <p>Table S-1 Safety System Parameters</p> <ul style="list-style-type: none">- Reactor power- RCS level- RCS pressure- Core exit T/C temperature- Level in at least one SG- Auxiliary feed flow in at least one SG | <p>RCS activity greater than Technical Specification allowable limits</p> <p>1 2 3</p> <p>SUA.1 RCS activity > Technical Specification Section 3.4.16 limits (Note 9) SUA.2 With letdown in service, letdown line area radiation monitor R-9 > 500 mR/hr (Note 9)</p> |
| | 5 RCS Leakage | None | None | None | <p>RCS leakage for 15 minutes or longer</p> <p>1 2 3 4</p> <p>SUS.1 RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. OR RCS identified leakage > 25 gpm for ≥ 15 min. OR Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min. (Note 1)</p> |
| 6 RPS Failure | None | <p>Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal</p> <p>1 2</p> <p>SS6.1 An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5% AND All actions to shut down the reactor are not successful as indicated by reactor power ≥ 5% AND EITHER:</p> <ul style="list-style-type: none">- Core Cooling RED Path entry conditions met- Heat Sink RED Path entry conditions met | <p>Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor</p> <p>1 2</p> <p>SA6.1 An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5% AND Manual trip actions taken at the RTGB are not successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8)</p> | <p>Automatic or manual trip fails to shut down the reactor</p> <p>1 2</p> <p>SU6.1 An automatic trip did not shut down the reactor as indicated by reactor power ≥ 5% after any RPS setpoint is exceeded AND A subsequent automatic trip or manual trip action taken at the RTGB is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8) SU6.2 A manual trip did not shut down the reactor as indicated by reactor power ≥ 5% after any manual trip action was initiated AND A subsequent automatic trip or manual trip action taken at the RTGB is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)</p> | |
| 7 Loss of Comm. | | <p>NOTES</p> <p>Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.</p> <p>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.</p> <p>Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.</p> <p>Note 4: The pre-calculated effluent monitor values presented in EALs RA1-1, RS1-1 and RG1-1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.</p> <p>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.</p> <p>Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is not required.</p> <p>Note 7: This EAL does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.</p> <p>Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.</p> <p>Note 9: Mode 1 applicable only when RCS temperature is ≥ 500°F</p> | None | <p>Loss of all onsite or offsite communications capabilities</p> <p>1 2 3 4</p> <p>SUT.1 Loss of all Table S-3 onsite communication methods OR Loss of all Table S-3 offsite communication methods OR Loss of all Table S-3 NRC communication methods</p> | |
| 8 CNMT Failure | | None | None | <p>Failure to isolate containment or loss of containment pressure control</p> <p>1 2 3 4</p> <p>SUB.1 ETHER: Any penetration is not isolated within 15 min. of a VALID containment isolation signal OR Containment pressure ≥ 10 psig with < one full train of depressurization equipment operating (one Containment Spray System train AND one Containment Cooling System train) per design for ≥ 15 min. (Note 1)</p> | |
| 9 Hazardous Event Affecting Safety Systems | None | <p>Table S-4 Hazardous Events</p> <ul style="list-style-type: none">- 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 | <p>Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode</p> <p>1 2 3 4</p> <p>SA9.1 The occurrence of any Table S-4 hazardous event AND EITHER:</p> <ul style="list-style-type: none">- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode | None | |
| F Fission Product Barriers | <p>FG1.1 Loss of any two barriers AND Loss or potential loss of third barrier (Table F-1)</p> | <p>FS1.1 Loss or potential loss of any two barriers (Table F-1)</p> | <p>FA1.1 Any loss or any potential loss of either Fuel Clad or RCS barrier (Table F-1)</p> | None | |

| Table F-1 Fission Product Barrier Threshold Matrix | | | | | | |
|--|---|--|--|---|--|---|
| | Fuel Clad Barrier | | Reactor Coolant System Barrier | | Containment Barrier | |
| | Loss | Potential Loss | Loss | Potential Loss | Loss | Potential Loss |
| A. RCS or SG Tube Leakage | None | None | 1. An automatic or manual ECCS (SI) actuation required by EITHER: <ul style="list-style-type: none">UNISOLABLE RCS leakageSG tube RUPTURE | 1. RCS leakage > capacity of a single charging pump (> 77 gpm) due to EITHER: <ul style="list-style-type: none">UNISOLABLE RCS leakageSG tube leakage 2. CSFST Integrity-RED Path entry conditions met | 1. A leaking or RUPTURED SG is FAULTED outside of containment | None |
| B. Inadequate Heat Removal | 1. CSFST Core Cooling-RED Path entry conditions met | 1. CSFST Core Cooling-ORANGE Path entry conditions met 2. CSFST Heat Sink-RED Path entry conditions met AND Heat Sink is required | None | 1. CSFST Heat Sink-RED Path entry conditions met AND Heat Sink is required | None | 1. CSFST Core Cooling-RED Path entry conditions met AND Restoration procedures not effective within 15 min. (Note 1) |
| C. Containment Radiation / RCS Activity | 1. Containment High Range Radiation Monitor R-32A or R-32B > 100 R/hr 2. Dose equivalent I-131 coolant activity > 300 µCi/gm | None | 1. Containment High Range Radiation Monitor R-32A or R-32B > 5 R/hr | None | None | 1. Containment High Range Radiation Monitor R-32A or R-32B > 2000 R/hr |
| D. Containment Integrity or Bypass | None | None | None | None | 1. Containment isolation is required AND EITHER <ul style="list-style-type: none">Containment integrity has been lost based on Emergency Coordinator judgmentUNISOLABLE pathway from Containment to the environment exists 2. Indications of RCS leakage outside of Containment | 1. CSFST Containment-RED Path entry conditions met 2. Containment hydrogen concentration ≥ 4% 3. Containment pressure ≥ 10 psig with < one full train of depressurization equipment operating (one Containment Spray System train AND one Containment Cooling System train) per design for ≥ 15 min. (Note 1) |
| E. EC Judgment | 1. Any condition in the opinion of the Emergency Coordinator that indicates loss of the Fuel Clad barrier | 1. Any condition in the opinion of the Emergency Coordinator that indicates potential loss of the Fuel Clad barrier | 1. Any condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier | 1. Any condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier | 1. Any condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier | 1. Any condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier |

| Table F-2 CSFST Entry Conditions | | | | |
|--|---|---|--|--------------------------------|
| Core Cooling RED Path | Core Cooling ORANGE Path | Heat Sink RED Path | RCS Integrity RED Path | Containment RED Path |
| Core exit T/Cs ≥ 1,200°F OR Both of the following: - Core exit T/Cs ≥ 700°F - Reactor Vessel water level ≤ Table F-3 thresholds | EITHER of the following: - Core exit T/Cs ≥ 700°F - Reactor Vessel water level ≤ Table F-3 thresholds | All S/Gs ≤ 9% (18%) AND Total FW flow to S/Gs is less ≤ 300 gpm or ≤ 0.2E6 pph | Temperature decrease in any RCS cold leg ≥ 100°F in last 60 min. AND Any RCS cold leg temperature is to the left of Limit A, Figure F-1 | Containment pressure ≥ 42 psig |

| Table F-3 Reactor Vessel Water Level Thresholds | | |
|---|--------------------|-------|
| RVLIS | No. RCPS Operating | Level |
| Full Range | None | 41% |
| Dynamic Head | 1 | 53% |
| | 2 | 35% |
| | 1 | 24% |

Figure F-1 Integrity Press.- Temp. Limit

The graph shows Temperature (°F) on the y-axis (500 to 350) and Pressure (psig) on the x-axis (0 to 300). A red shaded region labeled 'RED' is bounded by a red line (Limit A) and a vertical line at 200 psig. A green shaded region labeled 'GREEN' is to the right of 200 psig. A vertical line at 200 psig is labeled 'ECCS' and 'ECCS'.